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ALO-62
IMPROVING THE SAFETY
OF
LWR POWER PLANTS
FINAL REPORT

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IMPROVING THE SAFETY
OF
LWR POWER PLANTS

FINAL REPORT

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SECTION 1

INTRODUCTION - IMPROVING THE SAFETY OF LWR POWER PLANTS

1.1: STUDY DESCRIPTION

This report documents the results of the Study to identify current, potential research issues and efforts for improving the safety of Light Water Reactor (LWR) power plants. The Study was conducted by MHB Technical Associates of San Jose, California as authorized by Sandia Laboratories in Contract 13-7399.

This final report describes the work accomplished, the results obtained, the problem areas, and the recommended solutions. Specifically, for each of the issues identified in this report for improving the safety of LWR power plants, a description is provided in detail of the safety significance, the current status (including information sources, status of technical knowledge, problem solution and current activities), and the suggestions for further research and development. Further, the issues are ranked for action into high, medium, and low priority with respect to primarily (a) improved safety (e.g. potential reduction in public risk and occupational exposure), and secondly (b) reduction in safety-related costs (improving or maintaining level of safety with simpler systems or in a more cost-effective manner).

1.2: MODEL FOR SAFETY ASSESSMENT

LWR safety (risk) depends both on the probability and the consequences of an accident. In this Study, and in the U.S. Reactor Safety Study (WASH-1400),^{1/} risk is described as the product of accident probability and consequence multiplied together as shown in Figure 1-1. However, as discussed in Section 1.6, this mathematical process is not the only manner by which to describe risk. In contrast to the WASH-1400 method, policy makers in evaluating risks where the potential consequences of a major accident are very large, which is true for a nuclear accident, may wish to make their decision on the needs for improvements in reactor safety technology based primarily on the potential accident size alone.

1.3: SCOPE OF MHB STUDY

In this Study, the reports of Tasks I, II, and III are described in Sections 2, 3, 5, and 6 respectively. Specifically, in Section 2 of the report, a number of areas are identified for improving the design of nuclear plant structures, systems, and components in order to reduce the probability and/or magnitude of accidental and routine radioactive releases. Suggestions for mitigating the consequences to the public and workers to exposures from radioactive releases are described in Section 3 of the report (It is also important to note that safety improvements for reducing nuclear plant employee risk from routine, non-radioactive, industrial accidents are not addressed or evaluated in this report.) Section 4 reports on potential programs to improve safety assurance effectiveness; for example, the NRC licensing process and quality assurance programs, that could potentially improve the safety of LWR power plants. The criteria and methodology for evaluating and prioritizing the issues identified in Sections 2, 3, and 4 are discussed in Section 5, while a summary of this report, the conclusions, and the recommendations are described in Section 6.

The MHB Study, as shown in Figure 1-2, addresses accident prevention and consequence mitigation for both operating and planned LWR's. The emphasis of the MHB Study is on the safety aspects of the operating plants and those plants now under construction, since nuclear plants on the drawing board may not

be operational for nearly a decade. Thus, we believe that the risk for the next decade, and probably for the life of the United States' LWR program, will be clearly dominated by these operating plants and the plants now with construction permits. The MHB Study also encompasses the safety of LWR plant personnel as well as the members of the public.

FIGURE 1-2

SCOPE OF MHB STUDY

* PLANTS WITH AN NRC OPERATING LICENSE.

** PLANTS WITH AN NRC CONSTRUCTION PERMIT OR WITH AN APPLICATION SUBMITTED FOR A CONSTRUCTION PERMIT.

1.4: DOCUMENTS REVIEWED DURING STUDY

Numerous documents which address various aspects of the safety of LWR power plants were obtained and reviewed during the preparation of this report. In response to the accident which occurred at Three Mile Island, Unit 2 (TMI-2), on March 28, 1979, a number of reviews and studies have been initiated to identify the causes and lessons and effects of the accident. The President's Commission on the Accident at Three Mile Island has conducted public hearings, the most recent which we attended for Sandia in Washington, D.C. on August 21, 22, and 23 (for minutes of the meeting, see Appendix A of this Study). One of the authors of this report, Gregory Minor, is also a member of the review committee for the NRC's Special Inquiry on Three Mile Island headed by Mitchell Rogovin. Minor attended the first advisor's meeting during the week of August 13 to 17. NRC reports resulting from the NRC Staff's investigation of the TMI-2 accident which we reviewed as part of this Study include the Staff report on the generic assessment of feedwater transients of PWR's designed by the Babcock & Wilcox Company,^{2/} the TMI-2 lessons learned task force status report and short-term recommendations ^{3/} (for copies of the viewgraphs of Roger Mattson's June 25, 1979 presentation to the NRC Commissioners, see Appendix D of this Study), the investigation of the TMI-2 accident of the Office of Inspection and Enforcement,^{4/} the actions being taken by the

Office of Nuclear Reactor Regulation on licensee emergency preparedness,^{5/} the new Staff requirements for operator licensing,^{6/} and the Bulletins and Orders followup--primarily in the areas of auxiliary feedwater system reliability, loss of feedwater and small break loss-of-coolant analysis, emergency operating guidelines and procedures, and operator training.^{7/} On August 1, we attended, as part of the data gathering portion of this Study, the NRC briefing of reactor owners by Harold Denton on the short-term implication of TMI-2 (for Hubbard's minutes of this meeting, see Appendix B of this Study). In addition, we reviewed the information submitted by Pacific Gas and Electric Company responding to TMI-2 as part of the Diablo Canyon Station operating license proceeding,^{8/} and the information submitted by Public Service of Oklahoma responding to TMI-2 as part of the Black Fox Station construction permit proceeding.^{9/} Finally, we obtained and reviewed the Electric Power Research Institute (EPRI) report which analyzed the TMI-2 accident.^{10/}

The term "unresolved technical issues" identifies generic deficiencies which may contribute to amplifying the risk to public health and safety from LWR operation. Quantification of the resulting technical and economic risk, and the associated uncertainties, has remained an elusive goal. During this Study, a number of documents addressing "generic issues"

were reviewed. It appears that the "unresolved issues" may affect many operating or planned reactors. For example, 133 issues were designated in 1977 by the NRC as requiring Staff attention.^{11/} In January, 1979 the NRC further highlighted 17 of the items as having the highest safety impact.^{12/} Further, the NRC's Probabilistic Analysis Staff prepared a risk-based categorization of the issues during 1978.^{13/} These issues, each a potential deficiency affecting many commercial nuclear plants, have previously been identified in a variety of documents including the NRC's Technical Safety Activities Report (TSAR),^{14/} the Advisory Committee on Reactor Safeguards (ACRS) list of generic issues,^{15/} and finally, the list of issues raised by concerned NRC Staff members in response to the new NRC procedures for dealing with Staff dissent.^{16/ 17/}

Following the Three Mile Island Unit 2 accident, Harold Denton, in document SECY-79-344, briefed the NRC Commissioners on the Staff's plans to continue to perform these reviews and analyses necessary to complete generic tasks that address "unresolved safety issues" with minimum impact on current schedules. Denton reiterated that initially this task will include the generic tasks identified in NUREG-0510 that address "unresolved safety issues." However, several of these generic tasks will likely be expanded to address issues identified as a result of the TMI-2 accident. In addition, Denton believed

that new "unresolved safety issues" will likely be identified as a result of the TMI-2 accident. Thus, the "unresolved safety issues" task may be expanded to include generic tasks to address these new issues as they are identified. Denton concluded that the end products will be NUREG reports describing the Staff's evaluation of and conclusions for each issue. One can thus anticipate that the NRC's assessment of the lessons learned on Three Mile Island will result in a Staff re-evaluation of the scope and timing of the resolution of generic issues.

Another set of documents reviewed during this Study was WASH-1400 and the recent review of WASH-1400. WASH-1400 assessed the probability and consequence of a nuclear reactor accident. As such, WASH-1400 identified dominant contributors to plant safety. The report identified the uncertainty in their results as a factor of ± 5 in probability and a factor of ± 4 in prompt consequences, plus limiting the applicability of their results to 100 plants and a 5-year period.

The most recent review of WASH-1400 was conducted by the Risk Assessment Review Group, set up by the NRC and chaired by Dr. Harold Lewis. Their report, entitled Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission, NUREG/CR-0400, was published in September of 1978. The Risk Assessment Review Group held a dozen public meetings in 1977 and 1978 and received numerous presentations of data and

viewpoints, both supportive and critical of WASH-1400 methodology and results. The data presented at these meetings (several thousand pages) which was reviewed as part of this Study, represents one of the largest and most recent sources of information on reactor accident probability and consequences, covering a wide range of viewpoints and opinions. The RARG report does not quantify uncertainty, but does include the following observation:

"We are unable to determine whether the absolute probabilities of accident sequences in WASH-1400 are high or low, but we believe that the error bounds on those estimates are, in general, greatly understated."18/

In regard to the uncertainties in the accident consequences identified in WASH-1400, the Review Group made these observations:

"There is much disagreement about the details of the estimates made by the RSS team charged with making the disease and mortality estimates. For example, although all the members of the RSS team contributed their full and honest efforts to the task, the spectrum represented by that team was not broad enough to encompass the full range of scholarly opinion on the subject. This led the RSS team to make estimates with a narrower range of stated 'uncertainty' than would otherwise have been the case."19/

In 1978, one of the authors of this study, Dale Bridenbaugh, was a member of and consultant to the NRC research review group which participated in the development of the first annual NRC plan for research to improve the safety of LWR power plants. Five research topics that appear most likely to lead to risk reductions were identified as research projects.20/

In this Study, we have not conducted a detailed line-by-line assessment of the present and proposed LWR safety research programs and budgets of the NRC and DOE. Rather, recognizing that an independent view of LWR safety programs and priorities is being solicited, we have developed our recommendations free of the constraints inherent in existing budgets and research programs. However, in obtaining information for this report, we attended the ACRS Subcommittee meeting on Improved Safety Systems, June 26, 1979, where the ACRS reviewed the NRC and Department of Energy (DOE) program plans for research to improve LWR safety (the viewgraphs of the presentations to the ACRS Subcommittee are included as Appendix C of this Study).^{21/} The ACRS comments on the research program budget are documented in NUREG-0603 which was issued in July, 1979.^{22/} ACRS review and evaluations of the NRC's 1978 LWR safety research program was issued in December, 1978.^{23/}

The safety of operating plants is a key element in this Study. The U.S. NRC is conducting a Systematic Evaluation Program (SEP) to review eleven older BWR and PWR nuclear plants to determine the degree to which they comply with the current NRC regulatory practices for licensing new plants (Regulations, Regulatory Guides, Standard Review Plans, etc.). SEP is intended to provide basic information regarding the areas of strength and weakness of older plant designs compared to current regulatory

practices. Eventually, the SEP methodology is intended to provide the necessary information to enable the NRC to decide if any deficiency (or the sum) has a sufficient impact on safety to require backfitting of older plants. The program is also a first step toward eventual quantification of the aging effect on nuclear plant performance--a very desirable achievement. Documents describing SEP were reviewed as part of this Study.^{24/}

We have also reviewed a number of documents which discuss and evaluate the effectiveness of the NRC regulations and the NRC as regulators. For example, a recent General Accounting Office (GAO) report analyzed the effectiveness of the NRC's program for licensing nuclear power plant operators.^{25/} Numerous other reports have reviewed the effectiveness of the NRC's quality assurance program and its implementation.^{26/ 27/ 28/ 29/ 30/ 31/ 32/} The inability to adequately quantify quality assurance program effectiveness continues to introduce uncertainty into any risk-based quantification or categorization, particularly as related to the assignment of absolute values to reliability estimates.

1.5: NEED TO REASSESS LWR SAFETY PROGRAM PRIORITIES AND FOCUS

Past LWR safety research programs by the NRC and its predecessor, the AEC, have almost completely focused on accident prevention. Research has been generally directed towards

reducing the probability of core melt accidents, and particularly the probability of the large LOCA. One result is that over the years, the NRC and the nuclear industry have developed a mind set that "core melt accidents can't happen." As a corollary, little, if any, attention has been focused on mitigating the consequences of a fuel melt accident.

The accident at Three Mile Island, and the findings to date from the numerous investigations now underway, indicate that the priorities and assumptions that have formed the foundation of the LWR safety program must be reassessed. Consequence mitigation, as opposed to accident prevention, must have a larger role in the research program. Research program priorities must recognize that accidents can happen. A number of potential considerations in developing a framework for reassessing the priorities in the LWR safety research program are discussed in detail in Section 5 of this Study.

1.6: SAFETY IMPROVEMENT ASSESSMENT CRITERIA

The acceptability of LWR plant safety must also be judged, in part, by comparison with other risks. However, nuclear risks cannot easily be compared with other risks for their nature is different. For example, if risk is expressed as the product of probability X consequence, where:

$$\text{risk} \left[\frac{\text{deaths}}{\text{unit time}} \right] = \text{probability} \left[\frac{\text{accidents}}{\text{unit time}} \right] \times \text{consequence} \left[\frac{\text{deaths}}{\text{accident}} \right]$$

consider the example that two very dissimilar accidents can have the same average risk.

HIGH PROBABILITY - LOW CONSEQUENCE

$$1 \frac{\text{accident}}{\text{year}} \times 1 \frac{\text{death}}{\text{accident}} = 1 \frac{\text{death}}{\text{year}}$$

or

LOW PROBABILITY - HIGH CONSEQUENCE

$$\frac{1 \text{ ACCIDENT}}{10,000 \text{ YEARS}} \times \frac{10,000 \text{ DEATHS}}{1 \text{ ACCIDENT}} = 1 \frac{\text{DEATH}}{\text{YEAR}}$$

The public and government policy makers would generally view a 10,000 death accident as very undesirable, even if its average risk is low. Carried to the limit, one is faced with the "zero-infinity dilemma,"^{33/} where a near-zero probability times a near-infinite consequence is indeterminate and therefore a dilemma for the policy makers and the public.

Nuclear events may have low probability but can have very major consequences. It is possible that the effects of a major reactor accident may be so large as to prove unacceptable to the public and the policy makers. With a score of reactors in operation, the probability that a single large, but by no means the

largest, accident will occur within a few decades is not negligible. If the decision is made to continue to increase the United State's nuclear commitment, the policy makers and the public must understand their own personal risk related to that decision.

To some extent, risk reduction is possible. Improvement in reactor design, evacuation planning and practice, medical treatment capability, and siting can have a large impact on the safety of nuclear power. However, for those measures to be effective, it is essential that the policy makers and the public be fully informed of any major decisions in the areas listed.

This Study is not intended to address the question of the acceptability of the risk involved with the LWR reactors. The purpose is, rather, to compare LWR safety program options by conventional techniques so that the recommendations of this Study may be used as a tool in policy decisions by the government. Therefore, we proceed to the detailed discussions of accident probability reduction, accident consequence mitigation, and safety assurance effectiveness.

SECTION 1

REFERENCES

- 1/ WASH-1400 (NUREG--75/014), Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Washington, D.C., October, 1975. We refer herinafter to the study and the draft report as "WASH-1400."
- 2/ NUREG-0560, Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock and Wilcox Company, U.S. Nuclear Regulatory Commission, Washington, D.C., May, 1979.
- 3/ NUREG-0578, TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations, U.S. Nuclear Regulatory Commission, Washington, D.C., July, 1979.
- 4/ NUREG-0600, Investigation Into the March 28, 1979 Three Mile Island Accident by Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C., August, 1979.
- 5/ SECY-79-450, Memorandum to the NRC Commissioners on Licensee Emergency Preparedness, July 23, 1979.
- 6/ SECY-79-33-E, Memorandum to the NRC Commissioners on Operator Licensing, July, 1979.
- 7/ I&E Bulletin 79-08, NRC Bulletin and Orders Task Force Report, July, 1979.
- 8/ Report to Nuclear Regulatory Commission From Pacific Gas and Electric Company Describing Response Programs Following the Accident at Three Mile Island, July, 1979, NRC Docket Nos. 50-275 and 50-323.
- 9/ Response of Public Service Company of Oklahoma, Black Fox Station, Units 1 & 2, U.S. NRC Docket Nos. Stn. 50-556, 50-557 To NUREG-0578, I&E Bulletin 79-08, and Selected Issues on Emergency Preparedness, July 27, 1979.
- 10/ NSAC-1, Analysis of Three Mile Island - Unit 2 Accident, Nuclear Safety Analysis Center, EPRI, Palo Alto, California, July, 1979.

- 11/ NUREG-0410, NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Washington, D.C., January, 1978. (Note: NUREG-0371, entitled Approved Task Action Plans for Category A Generic Activities, is contained as Appendix F of NUREG-0410).
- 12/ NUREG-0510, Identification of Unresolved Safety Issues Relating to Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Washington, D.C., January, 1979.
- 13/ M. Cunningham, J. Murphy, M. Taylor, Preliminary Draft - Summary Report on a Risk Based Categorization of NRC Technical and Generic Issues, U.S. Nuclear Regulatory Commission, Washington, D.C., 1978
- 14/ Investigation of Charges Relating to Nuclear Reactor Safety, Hearings Before the Joint Committee on Atomic Energy, Vol. 2, 1976, U.S. Government Printing Office, Washington, D.C., pp. 1200 to 1445.
- 15/ Letter, M. Bender, ACRS Chairman, to Joseph Hendrie, NRC Chairman, entitled, "Status of Generic Items Relating to Light-Water Reactors: Report No. 6," dated November 15, 1977. (Note: Report #7 was issued by the ACRS on March 21, 1979.)
- 16/ NUREG-0138, Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director, NRR to NRC Staff, U.S. Nuclear Regulatory Commission, Washington, D.C., November, 1976.
- 17/ NUREG-0153, Staff Discussion of Twelve Additional Technical Issues Raised by Responses to November 3, 1976 Memorandum from Director, NRR to NRC Staff, U.S. Nuclear Regulatory Commission, Washington, D.C., December, 1976.
- 18/ NUREG/CR-0400, Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission, U.S. Nuclear Regulatory Commission, Washington, D.C., September, 1978, p. vii.
- 19/ Ibid 18, p. 23.
- 20/ NUREG-0438, Plan for Research to Improve the Safety of Light-Water Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Washington, D.C., April 12, 1978.
- 21/ Transcript of the June 26, 1979 Meeting of the ACRS Subcommittee on Improved Safety Systems, U.S. Nuclear Regulatory Commission, Washington, D.C.

- 22/ NUREG-0603, ACRS Comments on the NRC Safety Research Program Budget, Advisory Committee on Reactor Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C., July, 1979. Also see letter, Max W. Carbon, Chairman, ACRS to Joseph M. Hendrie, Chairman, NRC, entitled, "Studies to Improve Reactor Safety," dated August 14, 1979.
- 23/ NUREG-0496, 1978 Review and Evaluation of The Nuclear Regulatory Commission Safety Research Program, Advisory Committee on Reactor Safeguards, U.S. Nuclear Regulatory Commission, Washington, D.C., December, 1978.
- 24/ NUREG-0485, Status Summary Book, Vol. 1, No. 2, U.S. Nuclear Regulatory Commission, Washington, D.C., November 17, 1978.
(Note: Often called the "Brown Book.")
- 25/ EMD-79-67, Letter report on NRC's Program for Licensing Nuclear Power Plant Operators, U.S. General Accounting Office, Washington, D.C., May 15, 1979.
- 26/ NUREG-0321, A Study of the Nuclear Regulatory Commission Quality Assurance Program, U.S. Nuclear Regulatory Commission, Washington, D.C., August 1977.
- 27/ NUREG-0397, Revised Inspection Program for Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Washington, D.C., March, 1978.
- 28/ NUREG-0425, NRC Inspection Alternatives, U.S. Nuclear Regulatory Commission, Washington, D.C., February, 1978.
- 29/ EMD-077-30, Allegations of Poor Construction Practices on the North Anna Powerplants, U.S. General Accounting Office, Washington, D.C., June 2, 1977.
- 30/ Browns Ferry Nuclear Plant Fire, Hearings before the Joint Committee on Atomic Energy, Washington, D.C., September, 1975.
- 31/ NRC Press Release 76-122, Independent Assessment of NRC Quality Assurance Activities Planned, U.S. Nuclear Regulatory Commission, Washington, D.C., May 25, 1976
- 32/ EMD-78-80, The Nuclear Regulatory Commission Needs to Aggressively Monitor and Independently Evaluate Nuclear Power Plant Construction, U.S. General Accounting Office, Washington, D.C., September 7, 1978.
- 33/ Holdren, John, "Zero-Infinity Dilemma In Nuclear Power," presented at AAAS Meeting, Boston, Massachusetts, February 21, 1976.

SECTION 2

SAFETY IMPROVEMENTS BY REDUCING THE PROBABILITY OF ACCIDENTS

2.1: INTRODUCTION

Improvements in LWR safety can be achieved by reductions in either the probability of accidents or the consequences of accidents which have occurred. This section discusses possibilities for reducing the probability of serious accidents. Section 3 will discuss consequence reduction.

The most extensive probabilistic analysis of reactor accidents is in the Reactor Safety Study, WASH-1400. The WASH-1400 model is of interest because it identifies several areas where probability reductions could improve safety.

Figure 2-1 shows a breakdown of the model used in quantifying probability of reactor accidents leading to radioactive releases. The first four steps of the model involve the definition of accident initiating events and accident sequences including failure modes in the containment and common-cause failures. Steps 5-7 of the process establish numerical values of the different component and system failure rates based on actual or comparable data.

Radiation releases are quantified into five (9 for PWRs) categories covering the quantity of radiation released from the fuel

FIGURE 2-1
SUBTASKS IN QUANTIFYING THE
PROBABILITY OF RADIOACTIVE RELEASES

and the quantity released from the reactor during an accident. This is shown as Step 8 in Figure 2-1.

Finally, the component and system failures are combined into the failure rates of particular sequences and these are then combined into failure rates for the release categories to which they contribute (Step 9). Mathematically, this sequence of quantification steps is shown in Figure 2-2.

The result of the quantification is a set of probabilities for each release category plus an uncertainty bound or Error Factor (EF). These probabilities are a primary input to the calculation of total risk to the public.

A program to reduce risk (improve safety) may focus on any or all of the initiator or failure mechanisms as well as the human factors which may complicate otherwise benign events. However, not all programs to resolve safety issues get funded and/or resolved.

Many issues remain identified but unresolved.

Several vendor and regulatory bodies have compiled lists of unresolved safety issues with which they are familiar. Although not all of these issues are concerned with reducing the probability of an accident, they provide a good view of the type of component, system and design weaknesses which may effect probability.

2.2: SAFETY ISSUES IDENTIFIED BY THE ACRS

Evidence of possible design inadequacy of U.S. light-water reactors is accumulating in the form of "unresolved issues" which may effect the probability of accidents in operating or planned reactors. A total of 133 issues have recently been designated by the NRC as requiring staff attention.^{1/} Many of these issues have previously been identified in a variety of documents including the NRC's own Technical Safety Activities Report (TSAR) ^{2/}, the Advisory Committee on Reactor Safeguards (ACRS) list of generic issues ^{3/}, an internal assessment of the technical and business risk facing General Electric in fulfilling its orders for BWR's entitled the "Reed Task Force Report," ^{4/} and finally, the list of issues ^{5/} ^{6/} raised by concerned NRC staff members in response to an NRC procedure for dealing with staff dissent.

The term "unresolved safety issues" identifies deficiencies which may contribute to amplifying the risk to public health

and safety from power reactor accidents. The history of the U.S. nuclear power program has shown that, in many cases, the accidents which occur are not the ones which have been analyzed in the licensing process. However, after the fact, they are often found to have been on someone's list of safety concerns.

The NRC's Advisory Committee on Reactor Safeguards has been publishing since December 18, 1972, a series of lists of generic items (problems) which are of concern to light-water reactors. The November 15, 1977 list includes a total of 76 items of which 28 are considered unresolved.^{7/} Issues listed include such important and fundamental safety aspects as containment, pressure vessel, ECCS components, piping and electrical supplies.

The 1977 ACRS list of unresolved issues is shown in Table 2-1. Ten of the problems have been on the list since its inception in 1972, yet remain unresolved. Even when items are declared resolved, there is no certainty that a solution has been implemented. Anticipated Transients Without Scram (ATWS) represents an example of this category in that it has been declared resolved in an administrative sense. Forty-eight "resolved" items exist; it is not known what percentage of them resulted in real fixes in hardware or structures and, conversely, how many were "administrative" only.

TABLE 2-1

ACRS GENERIC ISSUES-RESOLUTION PENDING

3/

ACRS GENERIC ITEM	RELEVANT TO		PRIORITY FOR	
	PWR	BWR	ACRS	RESOLUTION NRC
GROUP II: (Resolution Pending Since December 18, 1972)				
1. Turbine Missiles	X	X	A	A-37, A-32
2. Containment Sprays	X		B	C-10
3. Pressure Vessel Failure By Thermal Shock	X		A	A-11
4. Instruments to Detect (Severe) Fuel Failure	X	X	C	-
5A. Excessive Vibration	X	X	B	-
5B. Loose Parts Monitoring	X	X	B	B-60
6. Non-Random Multiple Failures	X	X	A	C-13
6A. Reactor Scram Systems	X	X	A	A-9
6B. Alternating Current Sources Onsite & Offsite	X	X	A	A-35, B-56, B-57
6C. Direct Current Systems	X	X	A	A-30
7. Behavior of Reactor Fuels Under Abnormal Conditions	X	X	A	B-22
8. BWR Recirculation Pump Overspeed During LOCA		X	B	B-68
9. Seismic Scram	X	X	C	D-1
10. ECCS Capability for Future Plants	X	X	A	D-2
GROUP II A: (Resolution Pending Since February 13, 1974)				
1. Ice Condenser Containments	X		B	B-54
2. PWR Pump Overspeed During a LOCA	X		B	B-68
3. Steam Generator Tube Leakage	X		A	A-3, A-4, A-5
4. ACRS/NRC Periodic 10-year Review	X	X	C	Policy
GROUP II B: (Resolution Pending Since March 12, 1975)				
1. Computer Reactor Protection System	X		B	A-19
2. Qualification of New Fuel Geometry	X	X	C	B-22
3. BWR Mark III Containments		X	B	A-39, B-10
4. Stress Corrosion Cracking in BWR Piping		X	B	Policy
GROUP II C: (Resolution Pending Since April 16, 1976)				
1. Locking Out of ECCS Power Operated Valves	X	X	B	B-8
2. Design Features to Control Sabotage	X	X	A	A-29
3A. Decontamination	X	X	B	A-15
3B. Decommissioning	X	X	B	B-64
4. Vessel Support Structures	X		B	A-2
5. Water Hammer	X	X	A	A-1
6. Maintenance and Inspection	X	X	B	B-34
7. BWR Mark I Containments		X	A	A-6, A-7, A-39
GROUP II D: (Resolution Pending Since February 24, 1977)				
1. Interfaces	X	X	A	Policy, A-17
2. Capability of Hermetic Seals	X	X	C	C-1
GROUP II E: (Resolution Pending Since November 15, 1977)				
1. Soil-structure Interaction	X	X	C	A-40, A-41

2.3: SAFETY ISSUES IDENTIFIED BY THE NRC

The ACRS list is by no means a complete accounting of "unresolved safety issues." The NRC in 1974 tabulated the unresolved safety issues into an internal Technical Safety Activities Report (TSAR). When released in 1975 the TSAR listed 223 items of concern to the NRC of which 173 were categorized as having "an important impact on the licensing review process." ^{8/} The NRC apparently discontinued the TSAR but in the fall of 1976, in response to a memorandum by Rusche of the NRC, one or more members of the NRC staff identified 27 technical issues as "problems whose priority, progress, or resolution was, in their opinion unsatisfactory." ^{9/} The list of unresolved items identified by concerned engineers within the NRC is summarized in Table 2-2.

In October, 1976 the NRC Commissioners directed the NRC staff to develop a program plan for the timely resolution of the generic technical issues. Implementation of the NRC program began in April, 1977. Initially, each of the four NRC divisions reporting to the Office of Nuclear Reactor Regulation described and proposed those generic items it considered to warrant the highest priority. Proposals were received for 355 tasks ^{10/} of which, following consolidation and elimination, 133 tasks were eventually selected for review. A set of uniform criteria was applied to the generic technical activities to establish their

TABLE 2-2
NRC STAFF - SAFETY ISSUES OF CONCERN

ISSUE	RELEVANT TO		PRIORITY FOR RESOLUTION (NRC)
	PWR	BWR	
1. Treatment of Non-Safety Grade Equipment in Evaluations of Postulated Steam Line Break Accidents	X		A-22
2. Lack of Independence of Interlocks on ECCS Valves	X	X	Policy
3. Acceptability of Swing Bus Design of BWR-4 Plants (a)		X	Implementation
4. Loss of Off-Site Power Subsequent to Manual Safety Injection Reset Following a LOCA	X		Implementation
5. Analysis of Postulated Reactor Coolant Pump Rotor Seizure Incidents	X	X	Implementation (2 loop plts)
6. Protection Against Single Failures in Reactivity Control	X	X	B-3
7. Passive Failures Following a Loss of Coolant Accident	X	X	B-58, C-7
8. Probabilistic Assessment of Reliability	(Breeder Only)		--
9. Frequency Decay	X		A-35, B-70
10. Grid Stability	X	X	A-35
11. Interpretation of GDC-19 Control Room	X	X	--
12. Load Break Switch/Generator Breaker	X	X	B-53
13. Instrument Trip Setpoints and Standard Specification	X	X	Implementation
14. Computer Protection System	X		A-19
15. Overpressurization	X		A-26
16. Automatic Resetting of Reactor Trip System (RTS) Trip Bistable Relays	X	X	--
17. Passive Mechanical Valve Failures	X	X	B-58
18. Electrical Cable Penetrations of Reactor Containment	X	X	B-9
19. Analysis of the Interaction of Structures and the Supporting Soil	X	X	A-40, A-41
20. Completeness of the Review of Plants Referencing RESAR-3	X		Policy
21. Instruments for Monitoring Both Radiation and Process Variables During Accidents	X	X	A-34
22. Safety Implications of Control System Failures and Plant Dynamics	X	X	A-17
23. Effects of Steam Generator Tube Degradation on the Consequences of a Steam Line Break Accident	X		A-3, A-4, A-5
24. Improving the Availability of Offsite Power	X	X	A-35
25. Environmental Qualification and/or Re-Qualification of Safety-Related Equipment for a Steam Line Break Accident	X		A-21
26. Improvement of BWR Shutdown Reactivity Performance (PRT)		X	--
27. Electromagnetic Pulse Effects of a High Altitude Explosion of a Nuclear Weapon on Safety-Related Equipment of Nuclear Power Plants	X	X	--

(a) Also applies to BWR-3 plants.

priority for resolution. The 41 tasks shown in Table 2-3 were classified as Category A, warranting priority attention. The remaining issues were divided into 72 Category B tasks, 17 Category C tasks and 3 Category D tasks as shown in Table 2-4.

The lists of issues represent a shopping list for needed research as seen by the NRC staff and/or the ACRS. Most of the Category A items of the NRC's generic issues list are being funded for internal and external research. However, it is important to keep in mind that these were the high priority items at the time of the evaluation; namely, 1977/1978. Since then, new safety information has changed the priorities somewhat, as is discussed in the next section.

2.4: SAFETY ISSUES RESULTING FROM TMI-2 ACCIDENT

The accident at TMI-2 sent shock waves through the entire nuclear industry as well as the NRC. The safety implications were grave reminders of the basic fallibility of human processes of design, regulation, maintenance and operation. It is not surprising, then, that there have been numerous reviews, commissions, inquiries and recommendations. Within the NRC there have been studies of feedwater transients by NRR^{11/}, a study of the operational and radiological aspects of TMI by I&E^{12/}, a "Lessons-Learned" report by DOR^{13/}. In addition, there has been an accident analysis by EPRI^{14/}, the President's Commission headed by Professor Kemeny (their report is due October 25, 1979)

TABLE 2-3
CATEGORY A TECHNICAL ACTIVITIES

TASK NO.	TITLE:	RELEVANT TO	
		PWR	BWR
A-1	Water Hammer	X	X
A-2	Asymmetric Blowdown Loads on the Reactor Vessel	X	X
A-3	Westinghouse Steam Generator Tube Integrity	X	
A-4	Combustion Engineering Steam Generator Tube Integrity	X	
A-5	Babcock & Wilcox Steam Generator Tube Integrity		X
A-6	Mark I Short Term Program		X
A-7	Mark I Long Term Program		X
A-8	Mark II Program	X	X
A-9	ATWS	X	X
A-10	BWR Nozzle Cracking		X
A-11	Reactor Vessel Materials Toughness	X	X
A-12	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports	X	X
A-13	Snubbers	X	X
A-14	Flaw Detection	X	X
A-15	Decontamination	X	X
A-16	Steam Effects on BWR Core Spray Distribution		X
A-17	Systems Interaction in Nuclear Power Plants	X	X
A-18	Pipe Rupture Design Criteria	X	X
A-19	Digital Computer Protection Systems		Plants with digital computers.
A-20	Impacts of Coal Fuel Cycle		Environmental
A-21	Main Steam Line Break Inside Containment	X	
A-22	PWR Main Steam Line Break - Core and Primary Coolant Boundary Response (MSLB Outside Containment)	X	
A-23	Containment Leak Testing	X	X
A-24	Qualification of Class IE Safety-Related Equipment	X	X
A-25	Nonsafety Loads on Class IE Power Sources	X	X
A-26	Reactor Vessel Pressure Transient Protection (Overpressure)	X	
A-27	Reload Application Guide	X	X
A-28	Increase in Spent Fuel Storage Capacity	X	X
A-29	Design Features to Control Sabotage	X	X
A-30	Adequacy of Safety-Related DC Power Supplies	X	X
A-31	RHR Shutdown Requirements	X	X
A-32	Evaluation of Overall Effects of Missiles	X	X
A-33	NEPA Reviews of Accident Risks		Environmental
A-34	Instruments for Monitoring Radiation and Process Variables During Accidents	X	X
A-35	Adequacy of Offsite Power Systems	X	X
A-36	Control of Heavy Loads Near Spent Fuel	X	X
A-37	Turbine Missiles	X	X
A-38	Tornado Missiles	X	X
A-39	Determination of Safety Relief Valve (SRV) Pool Dynamic		X
A-40	Seismic Design Criteria - Short Term Program	X	X
A-41	Seismic Design Criteria - Long Term Program	X	X

TABLE 2-4a

CATEGORY B TECHNICAL ACTIVITIES

<u>TASK NO.:</u>	<u>TITLE:</u>	<u>RELEVANT TO:</u>	
		<u>BWR</u>	<u>SWR</u>
B-1	Environmental Technical Specifications	X	X
B-2	Forecasting Electricity Demand	X	X
B-3	Event Categorization	X	X
B-4	ECCS Reliability	X	X
B-5	Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments	X	X
B-6	Loads, Load Combinations, Stress Limits	X	X
B-7	Secondary Accident Consequence Modeling		X
B-8	Locking Out of ECCS Power Operated Valves	X	X
B-9	Electrical Cable Penetrations of Containment	X	X
B-10	Behavior of BWR Mark III Containment	X	
B-11	Subcompartment Standard Problems	X	X
B-12	Containment Cooling Requirements (Non-LOCA)	X	X
B-13	Narvik Test Data Evaluation	X	X
B-14	Study of Hydrogen Mixing Capability in Containment Post-LOCA	X	X
B-15	CONTEMPT Computer Code Maintenance	X	X
B-16	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	X	X
B-17	Criteria for Safety-Related Operator Actions	X	X
B-18	Vortex Suppression Requirements for Containment Sumos		X
B-19	Thermal-Hydraulic Stability	X	X
B-20	Standard Problem Analysis	X	X
B-21	Core Physics	X	X
B-22	LWR Fuel	X	X
B-23	LMFBR Fuel	LMFBRs	
B-24	Seismic Qualification of Electrical and Mechanical Equipment	X	X
B-25	Piping Benchmark Problems	X	X
B-26	Structural Integrity of Containment Penetrations	X	X
B-27	Implementation and Use of Subsection NF	X	X
B-28	Radionuclide/Sediment Transport Program	X	X
B-29	Effectiveness of Ultimate Heat Sinks	X	X
B-30	Design Basis Floods and Probability	X	X
B-31	Dam Failure Model	X	X
B-32	Ice Effects on Safety-Related Water Supplies	Reactors in Northern United States	
B-33	Dose Assessment Methodology	X	X
B-34	Occupational Radiation Exposure Reduction	X	X
B-35	Confirmation of Appendix I Models for "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors"	X	X
B-36	Develop Design, Testing and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units for Engineered Safety Feature Systems and for Normal Ventilation Systems	X	X
B-37	Chemical Discharges to Receiving Waters	X	X
B-38	Reconnaissance Level Investigations	X	X

TABLE 2-4b

B-39	Transmission Lines		
B-40	Effects of Power Plant Entrainment on Plankton	X	X
B-41	Impacts on Fisheries	X	X
B-42	Socioeconomic Environmental Impacts	X	X
B-43	Value of Aerial Photographs for Site Evaluation	X	X
B-44	Forecasts of Generating Costs of Coal and Nuclear Plants	X	X
B-45	Need for Power - Energy Conservation	X	X
B-46	Costs of Alternatives in Environmental Design	X	X
B-47	Inservice Inspection of Supports Class 1, 2, 3 and MC Components	X	X
B-48	DWR Control Rod Drive Mechanical Failures	X	
B-49	Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments	X	X
B-50	Post-Operating Basis Earthquake Inspection	X	X
B-51	Assessment of Inelastic Analysis Techniques for Equipment and Components	X	X
B-52	Fuel Assembly Seismic and LOCA Responses	X	X
B-53	Load Break Switch	X	X
B-54	Ice Condenser Containments		X
B-55	Improved Reliability of Target-Rock Safety-Relief Valves		X
B-56	Diesel Reliability	X	X
B-57	Station Blackout	X	X
B-58	Passive Mechanical Failures	X	X
B-59	N 1 Loop Operation in BWRs and PWRs	X	X
B-60	Loose Parts Monitoring Systems	X	X
B-61	Analytically Derived Allowable ECCS Equipment Outage Periods	X	X
B-62	Re-Examination of Technical Bases for Establishing GLs, LSSs, and Reactor Protection System Trip Functions	X	X
B-63	Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary	X	X
B-64	Decommissioning of Reactors	X	X
B-65	Iodine Spiking	X	X
B-66	Control Room Infiltration Measurements	X	X
B-67	Effluent and Process Monitoring Instrumentation	X	X
B-68	Pump Overspeed During a LOCA	X	X
B-69	ECCS Leakage Ex-Containment	X	X
B-70	Power Grid Frequency Degradation and Effect on Primary Coolant Pumps		X
B-71	Incident Response		
B-72	Health Effects and Life-Shortening from Uranium and Coal Fuel Cycles	X	X
B-73	Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel	X	X

TABLE 2-4c

CATEGORY C TECHNICAL ACTIVITIES

<u>TASK NO.:</u>	<u>TITLE:</u>	<u>RELEVANT TO:</u>	
		<u>BWR</u>	<u>PWR</u>
C-1	Assurance of Continuous Long-Term Integrity of Seals on Instrumentation and Electrical Equipment	X	X
C-2	Study of Containment Depressurization by Inadvertent Spray Operation to Determine Adequacy of Containment External Design Pressure	X	X
C-3	Insulation Usage Within Containment	X	X
C-4	Statistical Methods for ECCS Analysis	X	X
C-5	Decay Heat Update	X	X
C-6	LOCA Heat Sources	X	X
C-7	PWR System Piping		X
C-8	Main Steam Line Leakage Control Systems	X	
C-9	RHR Heat Exchanger Tube Failures	X	X
C-10	Effective Operation of Containment Sprays in a LOCA	X	X
C-11	Assessment of Failure and Reliability of Pumps and Valves	X	X
C-12	Primary System Vibration Assessment		X
C-13	Non-Random Failures	X	X
C-14	Storm Surge Model for Coastal Sites	X	X
C-15	NUREG Report for Liquid Tank Failure Analysis	X	X on coasts
C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection	X	X farm land
C-17	Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes	X	X

CATEGORY D TECHNICAL ACTIVITIES

D-1	Advisability of a Seismic Scram	X	X
D-2	Emergency Core Cooling System Capability for Future Plants	X	X
D-3	Control Rod Drop Accident	X	

and the NRC/TMI Special Inquiry Group under Mitchell Rogovin (report due December 31, 1979).

Each of these groups is coming up with its own list of recommendations. The earliest report to be completed was the NRC staff report on Feedwater Transients (NUREG-0560). Their recommendations are summarized in Table 2-5. In some cases their recommendations are mainly to prevent the recurrence of a TMI-2 accident (e.g. vessel level measurement, AFW 'auto-mation'). However, some of these and many of the staff's other recommendations have generic impact and point to the clear need for looking at reactor safety and licensing in a new light. Among the generic items are the need to improve the man-machine interface, the need for better safety reporting/data evaluation, better simulators and training, more analysis of the small break LOCAs and improved post-accident monitoring. Many of these same items also appear in reports by other groups.

The Lessons-Learned report focuses more on the immediate issues and changes which were discovered as a result of TMI-2. In many cases these items appear to be very parochial and restricted to PWRs in order to prevent a TMI recurrence. Others include more far-reaching effects. Examples of these two extremes are the need to provide a direct indication of PORV position and the need to inert all BWR Mark I and II containments.^{15/} The full list of short-term recommendations from

TABLE 2-5

NRR (NUREG-0560) RECOMMENDATIONS FOLLOWING TMI-2

1. TMI-2 changes per I&E bulletins.
2. Add instruments to detect subcooling of primary loop.
3. Study improved automation of some safety functions.
4. Study interaction of OTSG, ICS, PZR sizing.
5. Study OTSG transient handling capability.
6. Improve ability to cope with FW transients.
7. Improve auxiliary FW reliability.
8. Refine NRC criteria and requirements for process control.
9. Reevaluate control system role and safety implications.
10. Provide a more direct reading of vessel water level.
11. Decide proper balance between added automation and improved operator response.
12. Set criteria for equipment important-to-safety but not Class 1E.
13. Better indication of POR valve position.
14. Revamp NRC reporting and data-assembly processes.
15. Improve isolation of non-essential lines.
16. RHR design should be able to handle highly contaminated coolant.
17. Training operator's actions to focus on core cooling.
18. Extend defense-in-depth concept to include operator.
19. Consider augmenting operator with real-time, on-line analysis.
20. Consider expanding training beyond single failure criteria.
21. Improve PWR simulators to include flashing and ECCS failures.
22. Expand simulator training.
23. Increase refresh training rate for operators.
24. Expand procedures to include multiple failures (e.g. station blackout, reactivity anomalies, ATWS).
25. Allow for operator improvisation.
26. Improve layout of control rooms.
27. More emphasis on human factors.
28. Require more realistic FW transient analysis.
29. Extend size range of small break LOCA analyses.
30. Extend time of events considered in code analysis.
31. Improve NRC ability to analyze OTSG.
32. Revise standard review plan for transients.
33. Improve GDC regarding anticipated transients.
34. Technical specification changes to improve surveillance.

TABLE 2-6

RECOMMENDATIONS FOR SAFETY IMPROVEMENT (SHORT-TERM)
BY THE LESSON-LEARNED TASK FORCE

1. Emergency power supply requirements for the pressurizer heaters, power-operated relief valves and block valves and pressurizer level indicators in PWRs.
2. Performance testing for BWR and PWR relief and safety valves.
3. Direct indication of power-operated relief valve and safety valve position for PWRs and BWRs.
4. Instrumentation for detection of inadequate core cooling in PWRs and BWRs.
5. Diverse and more selective containment isolation provisions for PWRs and BWRs.
6. Dedicated penetrations for external recombiners or post-accident purge systems.
7. Inerting BWR containments.
8. Capability to install hydrogen recombiner at each light water nuclear power plant.
9. Integrity of systems outside containment likely to contain radioactive materials (engineered safety systems and auxiliary systems) for PWRs and BWRs.
10. Design review of plant shielding of spaces for post-accident operations.
11. Automatic initiation of the auxiliary feedwater system for PWRs.
12. Auxiliary feedwater flow indication to steam generators for PWRs.
13. Improved post-accident sampling capability..
14. Increased range of radiation monitors.
15. Improved in-plant iodine instrumentation.
16. Analysis of design and off-normal transients and accidents.
17. Shift supervisor's responsibilities.
18. Shift technical advisor.
19. Shift and relief turnover procedures.
20. Control room access.
21. Onsite technical support center.
22. Onsite operational support center.
23. Revised limiting conditions for operation of nuclear plants based upon safety system availability.

the Lessons-Learned task force is given in Table 2-6.

The EPRI report (NSAC-1, July, 1979) focuses on the sequence of events in the first volume, but they plan to add later supplements which "will concentrate on causes, lessons learned and generic remedial or preventative measures which may be appropriate." ^{16/}

In the I&E report (NUREG-0600) a great deal of attention is placed on evaluating the operational, procedural and radiological events against the regulations. They draw conclusion about the areas where regulations were violated ^{17/} but do not identify specific recommendations for reducing the probability of future accidents.

2.5: SAFETY ISSUES RECOMMENDED FOR RESEARCH

In the previous chapters of this section there have been hundreds of safety issues mentioned from sources mainly related to the regulatory functions (NRC, ACRS, DOR, NRR, I&E, etc.). Many of these are not considered in this Probability discussion since they deal with accident consequences - which assumes the accident has already occurred - and thus are more appropriate for Section 3. Of those remaining, many do not require research but, instead, need rule-making to decide on implementation. There are others which may be combined in areas warranting research to evaluate feasibility, alternative approaches, or potential for safety improvement. The following areas are considered

desirable areas of research with the goal of reducing the probability of accidents.

2.5.1: HUMAN FACTORS RESEARCH

The control rooms of today's nuclear power plants are generally antiquated and poorly designed from a human factors point of view. One person described them as based on 1950's concepts, designed in the '60's, built in the '70's, to be operated until the year 2000 or more. Each utility has design requirements which are unique for its systems and its control rooms. The result is vertical standardization by utility rather than horizontal standardization across the industry. This makes it difficult to standardize on a good human factors design and nearly impossible to have a simulator which closely resembles the many control rooms created for one generation of reactors. These deficiencies are clearly pointed out in the recent EPRI report, Human Factors Methods for Nuclear Control Room Design,^{18/} and are also mentioned to a lesser extent in the NUREG-0560 document.^{19/}

To compound the present absence of human factors design, nuclear control room suppliers are shifting to computer-based systems which are generally operated as a hybrid system in conjunction with the present control/display system. This step should be carefully researched, evaluated and justified as an

improvement before plants are allowed to operate on this principle. Presently these trial steps are being made without hard data to verify effectiveness and impact on safe operation. This implies the need for a human factors laboratory or simulator to evaluate the present systems and the incremental approaches to computer-based systems.

The TMI accident was originally laid at the feet of operator error. This blame was partially rescinded when it was discovered that the operator was working in a very difficult environment with less than desirable information. A human factors research facility could aid in correcting these shortcomings and also evaluate the effectiveness of control room concepts of the future before they are cast in concrete. This a high priority item.

2.5.2: AUTOMATION vs MANUAL PROCEDURES

Several of the recommendations of the TMI review groups are in conflict on this issue. The Feedwater Transient analysis (NUREG-0560) recommends on the one hand that greater automation of some safety features be studied while on the other hand they recommend that the operator's actions be considered an extension of the defense-in-depth concept. They further recommend studies on the balance between added automation and improved operator response. Such a study would be valuable in deciding what steps toward automation are a knee-jerk

reaction to TMI-2 and which are truly a safety improvement.

It will also help to decide at what point the operator becomes overloaded with additional data and procedures. (The NUREG-0560 recommends that procedures be extended to cover multiple failures such as ATWS; meanwhile, the President's Commission in the August 23, 1979 examination of Victor Stello, head of I&E, determined that the TMI operator was actually following a proper procedure for their plant readings but the wrong one to mitigate or prevent the accident. Proliferation of procedures may do more harm than good.)

Another major question this research could explore is the value of the "10-minute rule" which says that during the first 10 minutes of a transient or accident, the plant should be designed so that no operator action is required for activating safety functions. Is this really being adhered to? Some countries (Sweden and Germany) have gone to the "30-minute rule." Should the U.S. follow suit?

This program is also considered high priority.

2.5.3: ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS)

The ATWS event is potentially one of the most severe reactor accidents that could occur. It involves not only the decay heat but full power heat as well and can lead to severe breaches of containment. The ATWS controversy (now ten years old) has centered around how likely the event is to occur.^{20/} However, there is value in assuming it would occur and then

working to mitigate the effects, or working to reduce the probability of occurrence. The NRC Staff estimated the chances of a severe ATWS accident at four-in-seven between now and the year 2010.^{21/} This has been refined to a slightly lower probability (about a four-in-ten chance^{22/}) but is still far from an incredible event.

The research required is to evaluate the numerous alternatives for reducing the probability of an ATWS accident through redundancy, diversity, added systems, etc. versus the decision to focus on mitigating the consequences via improved release barriers, overpressure protection, etc. These alternatives are discussed in NUREG-0460 and in a condensed form in the article by A. Thadani (NRC) in Nuclear Safety.^{23/}

The urgency is mainly a result of the potential severity of the event if it should occur. The accident at TMI-2 has been declared a Class 9 accident by the NRC Staff.^{24/} This means it has exceeded the accidents analyzed in the licensing process and would have been classed as incredible prior to its occurrence. An ATWS accident has even more potential for destructive consequences, and therefore the public safety requires that it be prevented (or mitigated). An independent research program to provide a recommended solution may help break the ten year deadlock between the reactor vendors and the NRC.

2.5.4: ADVANCED SIMULATORS

The few simulators in use today to train reactor operators are relatively simple. This is not to say they are ineffective;

they are a large improvement over cookbook training, and their use should be increased. Presently their capabilities are somewhat limited. One recommendation of the Feedwater Transient Report (NUREG-0560) is that the simulator for PWRs be expanded to include capability of simulating boiling conditions, natural circulation, voiding and other conditions experienced in TMI-2. Such improvements in simulator sophistication will help to prepare operators to handle a wider range of upset conditions.

There is a need for research to improve models of abnormal operation or upset conditions which might possibly occur and provide the ability for the operator to train under these transient conditions. The result may be the need for more and bigger simulator facilities. Advanced simulators would also provide an opportunity to evaluate and study operator intuition and innovation in unusual circumstances. This research effort is considered high priority because it will predictably result in a short-term improvement in the safety of reactor operation.

2.5.5: "FLIGHT RECORDER" FOR A REACTOR

Most of the post-accident evaluations of the TMI accident sequence relied on the data from the B&W "reactimeter" as their most reliable data. The "reactimeter" is a multichannel recording device that was installed to measure the characteristics

of new B&W core designs during early operation. Fortuitously, it was operating at the time of the accident.

With the increasing size and complexity of nuclear plants, it becomes more and more difficult to evaluate exactly what happened during an accident or severe transient. The addition of a "flight-recorder" type of device in a control room or cable spreading room will provide valuable information for analysis of plant and operator actions during transient and accident conditions. Use of this information may help to indicate design deficiencies and reduce the recurrence of errors.

The research needed is to determine the optimal set of parameters to be measured and the safest, most reliable approach to measuring them. Implementation of this program is moderate priority.

2.5.6: REACTOR SITING ALTERNATIVES

The experience at Diablo Canyon nuclear plant has demonstrated the need for more restrictive and careful siting practices. It has cost years of delay and hundreds of millions of dollars to modify the existing reactor plant to try to upgrade it for the larger earthquake now known to be possible at the site. The larger earthquake potential was not known at the time the site was approved.

The focus of the NRC has been to establish safety restrictions which, if complied with, will permit operation of nuclear

reactors in potentially hazardous areas and sites. (See, for instance, generic items A-40, A-41, Table 2-3 and Item II-9, Table 2-1). This may be the wrong focus. Study efforts would be well spent to find suitable sites in non-seismic (or other hazardous condition) areas and ways to effectively distribute power into the seismic area. Clearly, such a program will require a trade-off between safety and jurisdictional considerations.

2.5.7: OTHER AREAS TO IMPROVE SAFETY BY REDUCING ACCIDENT PROBABILITY

The following are a collection of projects which are important but of lesser priority.

- POST-ACCIDENT MONITORING

The TMI-2 accident showed that installed instruments are inadequate to measure the effects of an accident. The NRC and ACRS have listed this as one of their concerns (See Table 2-1, item II-4 and Table 2-2 item 21). Improved post-accident monitoring may be able to limit the extent of an accident as well as help mitigate the consequences. The goal should be to devise information flow to the operator which will allow him to make the necessary decisions required to minimize the progress and damage of an accident. Research may be required to provide some of the instrument systems and/or ranges which are not now available. Priority of this program

is higher from a mitigation point of view than from a probability consideration.

- SAFETY/CONTROL INDEPENDENCE

The lack of independence of safety and control functions in some existing and planned reactors has contributed to concern on the part of the NRC (see Table 2-2, items 1 and 22) and has even contributed to some accidents and transients, but still the plants are designed with shared or interrelated safety/control functions.

For example, the HPI function at the TMI is used as make-up or control function but may also be called on as a safety function. The operator's action of turning it off manually should not have allowed its safety function to be compromised. Other reactor suppliers have also been identified as having weaknesses in their independence of safety and control functions.^{25/ 26/}

In several reactor accidents the reactor safety has been effected by a control system function, not a safety system. For example, Brown's Ferry relied on a condensate booster pump to provide cooling water, and TMI used the non-safety block valves to stop the small LOCA through the PORVs. On the other hand, control functions can complicate safety actions. An example is the Integrated Control System (ICS) on B&W plants whose

malfunction at the Rancho Seco plant caused a transient resulting in the OTSG's (the heat sinks) boiling dry.

The independence of safety and control functions needs to be studied as a potential for reducing the chance of accidents and transients which challenge the safety systems. Similarly, the control system functions which impact on safety need to be studied.

- PWR VESSEL LEVEL MEASUREMENT

Reactor vessel water level in a PWR is inferred from measurements of pressurizer level. Many of the post-TMI reviews have questioned this, but the Lessons-Learned is not specifically requesting a direct method of measurement. Because of the difficulty of a level measurement based on differential pressure measurement in the core vicinity, some research has been started in the area of alternative techniques to level measurements, such as sonic, time-domain-reflectometer (TDR), etc. This should be pursued and implemented to eliminate what is now an indirect source of data for assessment of a critical parameter.

- COMMON-CAUSE FAILURES AND THE SINGLE FAILURE CRITERIA

WASH-1400 made an attempt to quantify multiple failures due to a common cause. Their effort has been criticized as being less than complete in that it would take an

extraordinary knowledge and insight to evaluate all the possible common-cause failures and their impact on safety. The events of TMI involved a series of failures which may be classed as common-cause. As a result, the ACRS has recommended several studies to reevaluate common-cause failures and the effects of system interactions on safety.^{27/} Ultimately, the question must be answered in terms of revised regulations. Is the single-failure-criteria adequate for licensing? Is the present limited consideration of common-cause failures adequate? Research is needed to assess the potential safety improvement available by careful consideration and reductions in common-cause failures and to assess the adequacy of the present licensing criteria in these areas.

2.5.8 ONGOING RESEARCH INTO ACCIDENT PROBABILITY REDUCTION

The reader who is familiar with present research activities in the nuclear industry will find some very major programs missing from this list of recommendations. It is the intent of this Study to focus on additional areas which may not have received attention in the past, on existing areas needing additional efforts, and in some cases to reevaluate who should be doing the research. In any event, the major efforts underway in areas such as core cooling testing (LOFT), pipe crack studies, containment analyses and steam generator improvements should be considered valuable and necessary programs.

LIST OF REFERENCES

1. NUREG-0410, NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants, January, 1978, U.S. NRC, Washington, D.C., pages D-1 to D-7. (Note: NUREG-0371, entitled Approved Task Action Plans for Category A Generic Activities, Volume 1, Revision 1, is contained as Appendix F of NUREG-0410.)
2. "Investigation of Charges Relating to Nuclear Reactor Safety", Hearings Before the Joint Committee on Atomic Energy, Volume 2, 1976, U.S. Government Printing Office, Washington, D.C., pages 1200 to 1445.
3. Letter, M. Bender, ACRS Chairman to Joseph Hendrie, NRC Chairman, entitled, "Status of Generic Items Relating to Light-Water Reactors: Report No. 6," dated November 15, 1977. (Although an updated report was issued March 21, 1979, it has not been used for this study).
4. Ibid 2, Volume 1, page 47.
5. NUREG-0138, Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976, Memorandum from Director, NRC to NRC Staff, November, 1976, U.S. NRC, Washington, D.C.
6. NUREG-0153, Staff Discussion of Twelve Additional Technical Issues Raised by Responses to November 3, 1976, Memorandum From Director, NRC to NRC Staff, December, 1976, U.S. NRC, Washington, D.C.
7. Ibid 3.
8. Ibid 2, page 1203.
9. Ibid 1, page 7.
10. Ibid 1, page 7.
11. NUREG-0560 - Staff Report on the Generic Assessment of Feed-water Transients in Pressurized Water Reactors Designed by B&W Company, U.S. NRC, May, 1979.

12. NUREG-600 - Investigation into the March 28, 1979, Three Mile Island Accident by Office of Inspection and Enforcement, August, 1979.
13. NUREG-0578 - TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations, U.S. NRC, July, 1979.
14. NSAC-1 - Analysis of Three Mile Island-Unit 2 Accident, EPRI, Nuclear Safety Analysis Center, July, 1979.
15. Ibid 13, pages A-9 and A-19.
16. Ibid 14, Introduction (page unnumbered).
17. Ibid 12, pages I-B-1 through I-B-12 and pages II-F-1 through II-F-13.
18. Human Factors Methods for Nuclear Control Room Design, EPRI NP-1118-SY, June, 1979, Summary Report.
19. Ibid 113 page 8-7 through 8-9.
20. NUREG-0460 - Anticipated Transients Without Scram for Light Water Reactors, U.S. NRC, Vol. I and Vol. II, April, 1978, Vol. III, December, 1978.
21. Ibid, Vol. I, page 92.
22. Ibid 20, Vol. III, page F-6 and Appendix F, Figure 1 shows a total probability of core melt or excessive pressure reaching about $2 \times 10^{-2}/\text{yr}$ in 1990. For the period between 1990-2010 this would be about a 4 in 10 chance.
23. A. Thadani, E. Hagen, "Anticipated Transients Without Scram for Light Water Reactors, Nuclear Safety, Vol. 20, No. 4, August, 1979, pages 422-433.
24. NRC Staff Response to Board Question No. 4 Regarding the Occurrence of a Class 9 Accident at TMI, Salem 1, Docket No. 50-272, August 24, 1979, Barry H. Smith, Counsel for NRC Staff.
25. The Nugget File, Union of Concerned Scientists, R. Pollard editor, Jan. 1979, pp. 41, & 42. Describes problems where interconnected power monitoring channels to provide a control signal also caused common-cause failures in the safety function of the four measurements. (Main Yankee has a Combustion Engineering reactor.)

26. Letter, Dr. Hanauer, NRC to E.G. Case, NRC, Aug. 18, 1977. Regarding a problem at the Westinghouse reactor at Zion, Illinois, Dr. Hanauer states, "Westinghouse designs are characterized by the large number and types of interactions between control systems and related safety systems. They think this is great. I think it is unsafe. This feud has been going on for years."
27. Letter, M. Carbon (ACRS) to J. Hendrie (NRC), August 14, 1979, "Studies to Improve Reactor Safety" (see particularly Carbon's items 2, 3, 4, & 10).

SECTION 3

CONSEQUENCE MITIGATION

3.1: GENERAL DESCRIPTION - CONSEQUENCE MODEL

One main lesson learned from the TMI-2 accident is that a broader range of safety issues should be addressed in the LWR safety research program. In the past, LWR safety research has been almost totally devoted to preventing accidents. As a result of the multiple sequence of events which led to the actual TMI-2 accident, and the Risk Assessment Review Group's inability to determine theoretically the absolute probabilities of accident sequences in WASH-1400,^{1/} it appears that any calculation of the probability of core melt accidents, and the resulting accident prevention measures, will contain substantial uncertainties. Therefore, LWR safety research must also be carried out to provide answers on how to mitigate the consequences of routine and accidental radioactive releases. This Section of the Study addresses programs for mitigating the consequences of radioactive releases.

The consequences of a major release of radioactivity into the atmosphere from a reactor melt-down accident could be manifold:

- Short-term (weeks to months) deaths from exposure to high doses of radiation;
- An increased incidence of cancer deaths and genetic defects in the population exposed to lower doses of radiation;

- Non-lethal radiation caused illnesses-- thyroid damage in particular;
- Contamination of land and property by radioactivity; and
- Contamination of water by radioactivity.^{2/}

The beginning point of the consequence calculation is the specification of the postulated accident in terms of the quantity of radioactive material that could be released to the environment, the amount of energy associated with the release, the duration of the release, the time of release after accident initiation, the warning time for evacuation, the elevation of the release,, and the probability of the accident occurrence. The description of the range of radioactive releases from a potential accident are described in WASH-1400. Data for five (5) potential BWR release categories and nine (9) potential PWR release categories are described.^{3/} The release data represents the basic input to the consequence model.

A schematic block diagram of the WASH-1400 consequence model is shown in Figure 3-1. Each block in Figure 3-1 represents a set of factors, a transfer function that is utilized in the calculation of the consequences of a radiation release. In this Section of the Study we will discuss the present state of knowledge for these factors, and finally, we will develop recommendations to reduce the present uncertainties in the factors.

FIGURE 3-1
BLOCK DIAGRAM - CONSEQUENCE MODEL ^{4/}

The following LWR safety research programs which address how to mitigate the consequences of routine and accidental radioactive releases are described in this section of the Study:

<u>Section</u>	<u>Safety Program Description</u>
3.2	Model Development - Site Specific Class 9 Accident Consequence Analysis Code
3.3	Soil and Liquid Pathway Analyses/Interdiction Techniques
3.4	Low-Level Waste Management
3.5	Land Contamination Criteria Development
3.5.1	Decontamination Methods and Potential Costs

<u>Section</u>	<u>Safety Program Description</u>
3.6	Emergency Response Planning
3.7	Radiation Exposure of Plant Employees

3.2: MODEL DEVELOPMENT - SITE SPECIFIC CLASS 9 ACCIDENT
CONSEQUENCE ANALYSIS CODE

Following the enactment of the National Environmental Policy Act (NEPA), the NRC issued guidance on the treatment of accidents in environmental reports of light water reactors in the form of a proposed Annex to 10 CFR Part 50, Appendix D. In that guidance (36 FR 2285, December 1, 1971) it is noted that consequences of accidents beyond the design basis (called Class 9 accidents) could be severe, but that the probability of their occurrence is so small that their environmental risk is extremely low.^{5/}

The Annex stated that the consequences of Class 9 accidents need not be analyzed and, accordingly, the NRC's NEPA environmental reviews have not included calculations of the consequences of Class 9 accidents. Rather, NRC environmental impact statements have discussed these accidents only in a qualitative sense by restating the conclusions in the proposed Annex and by briefly referencing the existence of a more quantitative analysis in WASH-1400.

Background information describing this problem, and past NRC practices, is summarized in the "Description of Problem" portion of NRC Task Action Plan A-33 which states, in part:

"In 1971, the AEC determined that, consistent with NEPA, the environmental assessments of requests for construction permits and operating licenses should include consideration of the possible impacts from accidents. An Annex to 10 CFR Appendix D was proposed which provided guidance to applicants in this regard. This guidance was included in Regulatory Guide 4.2 and has constituted, since 1971, the basis for the Staff reviews. Since 1971, a considerable number of 'realistic' accident assessments have been made. In substance, these reviews have uniformly shown that the risks associated with potential accidental releases are very low.

The approach in these assessments, typically is limited to preparation of a two page narrative summary that qualitatively describes accident probabilities and the rationale for concluding that accident risks are low and a one page table that provides numerical estimates of consequences of various categories of accidents (excluding Class 9 events). The approach to developing these consequence estimates also involves a largely simplistic analysis; minor adjustments are made from case to case (basically to account for variations in power level, exclusion boundary distance and population density). These numerical estimates are also limited to airpathway consequences.

The Staff's environmental statement by its nature is typically the only document concerning NRC license reviews that receives wide public and government agency attention and exposure even though other documents are circulated and available. It is evident from the comments received on the Staff's statements that the present approach does not adequately inform the public regarding the substance and depth of NRC's safety reviews nor adequately respond to public and various government agency questions dealing with the risks of accidents.

The Environmental Protection Agency (EPA) and the Department of the Interior (DOI) expressed the need for an improved treatment of accident risks and an expansion of the Staff assessments to include quantitative estimate of Class 9 accidents.

Beginning in 1973, in response to EPA concerns, the Staff augmented its assessments to discuss the then on-going Rasmussen Study as it related to Class 9 risks. For its part, EPA agreed that updating of the standard assessment was not warranted until after the Rasmussen Study results were made available. This dialogue was renewed in 1974, with EPA recommending that a generic environmental statement be prepared on accident risks. After extended discussions, the NRC Staff reiterated its 1973 commitment to update the standard assumptions in the proposed Annex A. As a precursor to this update, the Staff committed to an extension of the WASH-1400 study to include a more in-depth evaluation of Class 3-8 accidents and to further explore the significance of variations in site and plant design characteristics. The Department of the Interior has routinely suggested that more attention be given to the site risks associated with liquid pathway. In mid-1977, DOI and NRC Staff met to discuss the DOI's generic concerns. DOI was informed of the Staff's programs to augment the generic studies in WASH-1400, but no commitments were made to revise the current approach (which, as noted above, includes no discussion on the impacts of accidental releases to the liquid pathway)." 6/

In light of the criticisms of the WASH-1400 accident probabilities made by the Risk Assessment Review Group, the NRC Commissioners have concluded, in part, that ".....the Commission does not regard as reliable the Reactor Safety Study's numerical estimate of the overall risk of reactor accidents." (Page 3 of NRC Statement on Risk Assessment and The Reactor Safety Study Report In Light of the Risk Assessment Review Group Report.) Consequently, the NRC Staff is left with no theoretical basis to exclude the consideration of Class 9 accidents. In addition, the NRC Staff has concluded that the accident at TMI-2 was a Class 9 accident, even though the release of radioactive material

to the offsite population was very small.^{7/} Thus, in summary, the NRC does not have a technical basis, based on either operating experience or on a theoretical study, for excluding the environmental assessment of Class 9 accidents in reviewing sites.

The consequence model developed for WASH-1400, the CRAC Code, was developed to estimate aggregate societal risks and not to estimate site specific features. The CRAC model contains many simplifying assumptions that introduce substantial uncertainty into any site specific calculations. Safety research programs are recommended to complete and augment the programs presently underway to complete the necessary development of analytical models and corresponding experimental verification in order to reduce the uncertainties in the site specific application of the CRAC Code. In Section 3.2.1, we address and identify some of the uncertainties in the CRAC Code. In addition, we describe some of the sensitivity studies presently underway (the majority being conducted by Sandia as noted herein).

3.2.1: UNCERTAINTIES IN CRAC ACCIDENT CONSEQUENCE MODEL

Consequences of Class 9 reactor accidents have been previously modeled by the NRC with the CRAC model as described in Appendix VI of WASH-1400. The long-term consequences predicted in the draft of WASH-1400 for reactor accidents have been independently reviewed by a committee set up by the American Physical

Society,^{8/} but the short-term consequence predictions are only now being examined in great detail by an independent group (Sandia). The importance of an independent review is underscored by the errors the A.P.S. study group found in the long-term consequence predictions made in the draft of WASH-1400. Most of the A.P.S. corrections were accepted in the final WASH-1400 report.^{9/}

The U.S. Environmental Protection Agency (EPA) has also reviewed WASH-1400.^{10/} Its report contains information as to how to correct final WASH-1400's optimistic assumptions about latent cancer incidence. Additional technical discussions of both the long-term and the short-term impacts of a reactor accident are discussed in the Union of Concerned Scientists (UCS) Critique of WASH-1400.^{11/} Consequently, information already exists in the literature which can be used to estimate the long-term effects of accidents. (Where WASH-1400, EPA, the A.P.S., and the UCS report differ, such as in the extent of land contamination, or cancer incidence, the spread in numbers can be taken as a measure of uncertainty.)

To date, WASH-1400 represents the only substantial analysis of short-term reactor accident impacts. An independent, quantitative review is needed to ascertain whether or not there are errors contained in WASH-1400's prediction of

short-term fatalities similar to those found by the A.P.S. group in draft WASH-1400's treatment of long-term effects. The possibilities for error seem large because of the existence of a dose threshold, below which the probability of short-term fatalities are negligible. Relatively minor mistakes or questionable assumptions in accident simulations could conceivably have a significant effect on predictions. The Center for Environmental Studies at Princeton University has conducted a partial review of WASH-1400 along these lines for a Swedish reactor (Barsebäck). Based on the Princeton results, it appears that different assumptions can lead cumulatively to significantly larger short-term consequence frequencies at Barsebäck than would be calculated using the WASH-1400 model. For example, if one accepts the Princeton deposition and plume rise approach, if only minimal health treatment were available, and if evacuation were ineffective, then the increase in Barsebäck accident consequences could be as much as a factor of 30 greater than WASH-1400 results.^{12/} We recommend that such additional studies be initiated, and that the existing studies as described in the following, be concluded with a high priority in order that site specific Class 9 accident consequence analysis can be conducted for all existing and planned reactor sites.

The NRC, in a presentation to the Risk Assessment Review Group (RARG), also acknowledged the need for further study of the WASH-1400 consequence model. Dr. Wall, then Chief of the NRC's Probabilistic Analysis Branch (PAB), provided the RARG with a summary of the major technical comments on the final report of WASH-1400. The six commentators on the final document are : (a) EPA;^{13/14/15/} (b) Ford Foundation/MITRE Report - "Nuclear Power: Issues and Choices;"^{16/} (c) Department of the Interior;^{17/} (d) Frank von Hippel;^{18/19/} (e) Joel Yellin;^{20/21/} and (f) Henry Kendall.^{22/} Dr. Wall stated that only unresolved issues would be addressed in his presentation. Since a number of issues were expressed by more than one individual or organization, issues have been ordered by subject, with the appropriate commentators listed underneath. The PAB response to each comment is also included in the tables.^{23/} The critiques of the consequence model are summarized in Table 3-1.

TABLE 3-1
CONSEQUENCE MODELING
COMMENTS ON FINAL WASH-1400 24/

COMMENT:

PAB RESPONSE

-
1. WASH-1400 averaged population data from actual sites to form six composite sites. (b, d, e)	- Not totally accurate. WASH-1400 weighting of population sectors is judged to be adequate to represent peak and average consequences for 100 reactor-risk.
-
2. Correlations between wind direction and population distributions were not incorporated. (b, d, e)	- Correct. WASH-1400 objective emphasized risk from 100 reactors. It was judged that the use of uniform wind rose would be acceptable for the aggregate risk from 100 reactors.- In WASH-1400, single reactor risk equals 1% of 100 reactor aggregate risk. WASH-1400 inadequately explained that single reactor risk was for a 1000 MWe LWR at an average site. Specific sites were expected to vary substantially from this central estimate.- Subsequent correlation studies between population, weighted for health effects and wind rose show: • Minimal correlation at most sites.• Equal numbers of positive and negative correlations. Net effect on aggregate risk of 100 reactors is expected to be small.
-

TABLE 3-1

CONSEQUENCE MODELING (contd)

COMMENT:

PAB RESPONSE:

3. Averaging site-by-site consequences over all sites would be more accurate. (b)

- We question this observation.
- One year of hourly readings of meteorological data is unlikely to be available for all sites.
- Indian Point site had largest population sector and had positive correlation between wind direction and population. If all sites had equivalent correlation between their largest population and wind direction, 100-reactor risk would only increase about 10-fold. As stated in 2, above, overall impact on 100-reactor risk is therefore assumed to be small. (See Figure 3-2)
- Site-to-site latent risks:
 - Probability varies about factors of 8 from "average" site.
 - Maximum calculated consequences vary about factors of 4 from "average" site.

4. Temporal variation of wind direction is neglected. Multi-station meteorological data would be preferable. (b)

- Correct. Preliminary comparisons between unidirectional model and variable-direction model generally show WASH-1400 model to be slightly conservative.
 - Multi-station meteorological data is generally unavailable; prohibits use of more sophisticated model.
-

TABLE 3-1
CONSEQUENCE MODELING (contd)

COMMENT:

PAB RESPONSE:

5. Latent health effects modeling may not be adequate.

a. Recommendation to use average of BEIR relative and absolute risk models as central estimate. Factor of two increase in latent cancer fatalities. (a)	- Averaging of two risk models is scientifically indefensible. No consistent relationship between spontaneous cancer rates and additional radiation-induced cancer risk.
b. Departure from BEIR by using dose-effectiveness factors not justified. Further factor of two increase in latent cancer fatalities. (a, b)	- New experimental data supports lower estimate for low doses of low-LET radiation than BEIR. Draft NCRP-40 supports WASH-1400 approach, but ICRP-26 goes in opposite direction.
c. WASH-1400 applied dose-effective factors to all low-level radiation exposures whereas NCRP-40 recommends limiting their application to low-LET radiation. (b)	- Correct. However, high-LET radiation represented only 1% of dose, so error is small compared to overall uncertainties.
d. Experimental data published since WASH-1400 suggests that WASH-1400 genetic effects may be understated and have greater uncertainty. (b)	- Different interpretation of megamouse experiments at Oak Ridge Lab (factor of 2 increase).
- Different interpretation of Canadian data related to polygenic diseases (increases uncertainty but interpretation is controversial).
-

TABLE 3-1
CONSEQUENCE MODELING (contd)

COMMENT:

PAB RESPONSE:

6. Insufficient clinical data to support LD_{50/60} of 510 rads. (a)

- No general disagreement that:
LD_{50/60} is 340 rads when only minimal therapy is available.

Availability of supportive therapy would increase LD_{50/60}.
- Increasing LD_{50/60} from 340 to 510 rads as a result of supportive therapy based upon:
 - Judgment of experienced physicians in the treatment of severely irradiated patients.
 - Experiments with dogs.
 - Factor of 4 decrease.

7. WASH-1400 results indicate that most eventual latent cancer fatalities (i.e., ~ 83%) are a result of exposure during first week after accident. This points up need for rapid response in evacuation plans and inherent limitations in any long-term decontamination program. (b)

- WASH-1400 results were misleading since they were based upon BWR-1 release category which is atypical.
- Subsequent analyses including correction of programming error shows:

Release Category	% Latent Cancer Fatalities on the Average Due to 1st Week's Exposure
PWR-1	38
(BWR-1)	
PWR-2	23
- PWR-2 is more representative mixture of radionuclides.

FIGURE 3-2
SITE SPECIFIC CALCULATIONS 25/

Sensitivity studies of the WASH-1400 consequence model have been conducted by Sandia. Dr. P.E. McGrath, then of Sandia Laboratories, described the studies Sandia is conducting for the NRC's Probabilistic Analysis Staff to the RARG. Dr. McGrath stated that Sandia was taking a four-step approach to simplifying the consequence model: determine the limits of variables, set all other variables to obtain maximum sensitivity, set each variable alternately in turn to its maximum and minimum value, and inspect the results to determine whether the outputs were sensitive to the variable. Dr. McGrath added that there are three objectives to this study: identification of inputs that need to be better specified in order to more accurately assess the consequences of an accident; identification of the parts of the model that can be ignored, or left as approximation, without greatly affecting the accuracy; and identification of the parts of the model that need the highest accuracy. Dr. McGrath summarized their findings to date as follows: About 20 nuclides can be eliminated; use only Cs-137 for interdiction time; eliminate finite cloud correction factor at low concentration levels; eliminate LLI from early fatality calculations; each factor in dispersion calculation is important under certain assumptions; pay close attention to deposition velocity, washout coefficient and possibility of rain; could combine thermal stability classes A and B; and ignore release height for distances greater than 5 miles.^{26/} McGrath's

assessment of the uncertainties in the consequence model are tabulated in Table 3-2. Dr. McGrath noted that one cannot add all of these uncertainties. A first approximation would be to take the square root of the sum of the squares, resulting in an overall uncertainty factor of about 6.^{27/}

TABLE 3-2
UNCERTAINTIES IN CONSEQUENCE ESTIMATES ^{28/}

<u>MODEL AND DATA</u>	<u>UNCERTAINTY FACTOR</u>
● <u>EARLY FATALITIES</u>	
Accident Description	3 (Consequence)
Meteorology	5 (Consequence)
	2 (Probability)
Site Description	4 (Consequence & Probability)
Radiation Exposure	4 (Consequence)
● <u>LATENT CANCER FATALITIES</u>	
Accident Description	4 (Consequence)
Meteorology	"Small"
Site Description	"Small"
Radiation Exposure	4 (Consequence)

The following is a summary of some of Sandia's investigations to date (primarily by J.L. Sprung) of the adequacy of the meteorological transport model developed for WASH-1400.^{29/ 30/ 31/ 32/ 33/ 34/}

3.2.1.1: Accident Consequences - Plume Mixing Heights

Risk estimates for early fatalities, calculated using the Consequence Model (CRAC) of WASH-1400, are minimally sensitive to the specification of mixing heights. At distances less than 3 miles, conditional probabilities of early fatalities are increased by about a factor of 3 by decreasing mixing heights by a factor of 10. At longer distances, early fatalities are essentially unaltered. Substantial plume penetration of a low lying inversion layer would lead to a large decrease in early fatalities.^{35/}

3.2.1.2: Accident Consequences - Wind Shear

Explicit treatment of wind shear in CRAC would increase areas of plume coverage and decrease integrated plume exposures, which would cause both early fatalities and total latent cancer fatalities to decrease. However, the size of these decreases (less than a factor of 2) is not large enough to warrant modification of CRAC.^{35/}

3.2.1.3: Accident Consequences - Plume Jet Angle

Centerline heights of moisture-laden, radioactive plumes are unaffected by plume moisture content, atmospheric relative humidity, or radioactive decay. Plume jet angle and momentum

upon containment rupture can increase or decrease plume rise by up to 50 percent, but cannot be modeled unless jet angle can be specified. Continued use of the Briggs' formulas for plume rise is recommended.^{35/}

3.2.1.4: Accident Consequences - Rain

A hybrid rainstorm model, which reflects the spatial and temporal variability of rainstorms and assumes runoff at rates typical of urban areas, was incorporated in CRAC. Although consequences of single meteorological sequences containing rainstorms can be greatly altered by use of this model, aggregate calculations (91 meteorological sequences) are essentially unchanged, principally because few sequences contain rain.^{35/}

3.2.1.5: Accident Consequences - Site Specific Population Density

Site specific aggregate risk calculations were made for Indian Point and Zion. These calculations weighted the actual population distribution about each reactor site using the annual wind rose at the site. These calculations suggest that, if all of the 68 sites examined by WASH-1400 had nearby populations as large and as unfavorably situated as those of these two sites, the early fatality risk estimates of WASH-1400 would be increased by at most a factor of ~10. ^{35/}

3.2.1.6: Accident Consequences - Availability of Meteorological Records

Results from risk calculations repeated using disparate meteorological records suggest that adequate risk estimates for early fatalities can be obtained for sites, for which wind speed, atmospheric stability, and precipitation data are unavailable, by substituting data from another site, which has similar topography and meteorology. Substitute meteorology calculations suggest that the composite population distributions used in WASH-1400 caused the probabilities of early fatalities for low consequence events to be overestimated by about a factor of 3 and to be underestimated by about the same factor for higher consequence events, having probabilities of occurrence of $\sim 10^{-9}$ per year of reactor operation. The method used to construct composite population distributions introduced no significant error into the estimates of total latent cancer fatalities.^{35/}

3.3: SOIL AND LIQUID PATHWAY ANALYSES/INTERDICTION TECHNIQUES

Many nuclear power plants are located directly on the edges of significant water bodies, such as the Mississippi River system and the Great Lakes. Yet, little or no preparations have been made to interdict the flow of contaminated groundwater from beneath the reactor containment buildings into these bodies of water should a meltdown occur. Installing the necessary interdiction systems as a standard part of the safety equipment of each nuclear power plant should be evaluated as part of the LWR safety research program.

The WASH-1400 authors assumed that partial core melt always led to complete core melt. They further assumed, because of lack of direct experience, that once a core lost its initial configuration, it would be difficult to contain, i.e., even if the containment building above the core were not to fail, the core itself would probably melt through the concrete foundation. It is not clear that the core would necessarily follow this course. It may, instead, melt down but remain contained from below by the foundation due to adequate upward heat transfer. If the molten core has penetrated the containment building through the bottom, its interaction with the surrounding soil and water table was not well-defined. Furthermore, core melt-through was not evaluated with the same comprehensiveness as were atmospheric releases because of the latter's more immediately observable adverse effects.^{36/}

The WASH-1400 report indicates that airborne releases are much more significant contributors to the total risk than releases via the liquid pathway. The U.S. Department of Interior, as discussed briefly in Section 3.2 of this Study, did not agree with the notion that the liquid pathway was as insignificant as indicated in the WASH-1400 report and recommended additional study of the effects of variation of the hydro-geological conditions from site to site. The NRC accepted the comment and instituted a research program at Sandia (study results are scheduled to be

released prior to January 1, 1980). The WASH-1400 group assumed that there would be considerable scrubbing of fission products by the soil as the substances moved toward the surface; therefore, such retained fission products should be included in the liquid pathway analysis.^{37/}

In summary, models to assess the consequences of the possible massive contamination of water by radioactive releases, particularly strontium-90, on health, water supplies, costs, or interdiction schemes have not been adequately developed. This may be a significant omission when evaluating LWR safety. Since many nuclear plants are located near large bodies of water and these bodies of water are important sources of drinking water or food, the consequences of water contamination as a result of a reactor meltdown becomes a vital issue. As a result of comments by the American Physical Society Study Group on the draft WASH-1400 presentation^{38/} the calculated amount of strontium-90 released into the ground water has been increased by a factor of 1000 in the final WASH-1400. The total amount of strontium-90 which is calculated by WASH-1400 to leach into the ground water and subsequently the nearby water is now 150,000 curies (or about 3% of the total strontium-90 inventory) released over a period by about one year.^{39/} For comparison, to dilute the solution of strontium-90 inventory to equal the Maximum Permissible Concentrations (MPC) would require approximately one-half the annual discharge of the Mississippi River.^{40/}

WASH-1400 indicates that "the effects of contamination on water supplies has not been considered in detail," ^{41/} and assumes without a detailed analysis that streams and rivers would be contaminated for only a "short-time." ^{42/} The inadequate analysis of water contamination is a major flaw in WASH-1400 and the NRC's licensing process. The financial costs and the potential societal dislocation due to major water contamination from strontium-90 and other isotopes appear to be potentially very large. The analytical models of the soil and liquid pathway deserve continued development as part of the LWR safety research program.

3.4: LOW-LEVEL WASTE MANAGEMENT

The question of Fuel Cycle and High-Level Waste Management research are beyond the scope of this Study. Clearly, however, many studies of methods for high-level waste disposal have been conducted during the past 20 years. Equally clearly, the long-term disposal of high-level wastes remains an unresolved technical problem.

Turning to other than high-level wastes, however, as discussed by the ACRS in their recent assessment of the NRC's safety research program budget, the NRC faces immediate challenges relative to setting criteria for the disposal of low-level wastes, for the management of uranium mill tailings, and for the handling of spent fuel from commercial nuclear power plants. The ACRS concluded that the current NRC research program does not appear

to reflect the relative urgency with which each of these problems needs to be addressed. Attention to research problems associated with the handling, storage, and retrieval of spent fuel, for example, appear to the ACRS to be inadequate. In addition, the ACRS believed that the lack of clearly defined goals and specification as to how each research project fits into the overall program plan have resulted in a plethora of ongoing projects which are difficult to review and manage. The need for better coordination and direction is obvious.^{43/}

The ACRS recommendations for programs to improve the safety of low-level wastes appear equally applicable and consistent with the recommendations and conclusions of this Study. The following is a brief summary of the safety research programs, related to low-level wastes, that were recommended by the ACRS:

"Increased Attention to Low-Level Wastes

Data presented to the ACRS indicate that over 90% of the total volume of radioactive wastes being generated today are in the low-level category. Although these contain less than 1% of the total radioactivity to be handled, the volume of such wastes for calendar year 1977 was about 2.5 million cubic feet. Indications are that the handling of such wastes at the burial sites is largely done by hand (with associated high personnel doses). Because the waste containers are randomly placed in the trenches, space utilization is poor, voids exist, and retrievability would be difficult. Because of poor planning and management, several existing low-level waste disposal sites have had to be shut down and will represent continuing problems for years to come. Because of the lack of criteria for disposal of such wastes, commercial companies are continuing current operations and planning for the future without clear guidance on proper procedures. The ACRS urges that the

NRC address these problems as promptly as possible and provide the industry with the guidance it needs. To the extent that research is needed to solve certain aspects of these problems, it should be given top priority. These efforts should include increased research on methods, such as incineration and acid digestion, that can be used for reducing the volumes of the wastes to be handled, taking into account the active DOE programs in this area. Proposed FY 1981 allocations in this area (Item 10.b.) appear inadequate. In addition, no funds have been included in the proposed budget allocation for research on seeking 'Alternatives to Shallow Land Burial.' This is an important area that should be addressed in a meaningful way. Lastly, funds for research on the decommissioning and long-term care of low-level waste burial sites (Item 11.b.) appear to be totally inadequate (only one of four designated research project areas is to be funded and then only at a level of \$100 thousand.)

Ground Water Hydrology

Research on both high-level and low-level waste management includes items directed to ground water hydrology. The ACRS recommends that these efforts be closely coordinated with related studies on core melt accidents so as to assure maximum interchange of information.

Methods for Assaying Wastes

One area not apparently addressed by the current research program is the need for the development of equipment for assaying the radionuclide content of waste packages as received at waste disposal sites. Such equipment is necessary for determining whether wastes as received are within the overall radionuclide limits and whether they contain acceptable concentrations of the transuranics. No funds are allocated to these needs in the proposed FY 1981 budget. The Committee recommends that this situation be corrected.

Management of Gaseous Wastes - Fuel Cycle

NRC research on waste management is directed almost exclusively to the handling and disposal of liquid

and solid wastes. The Committee recommends that attention also be directed to research needs for the removal, confinement, and long-term storage of gaseous wastes. The experience at TMI, for example, showed that problems were encountered in the holdup system for gaseous waste releases. The proposed FY 1981 budget allocates \$150 thousand for research in this area (Item 10.a.). To the extent practicable, these problems should be more adequately addressed.

Emergency Planning

The proposed FY 1981 budget includes an item (10.c.) directed to research on 'Decorporation Techniques for Radionuclides.' The objective of this effort would be to evaluate methods for effecting the removal of radionuclides deposited internally. Unfortunately, this effort is not at present scheduled to be funded. The ACRS recommends that funds be provided for research in this area and that it be directed primarily to counter-measure actions in nuclear accidents." 44/

3.5: LAND CONTAMINATION CRITERIA DEVELOPMENT

The potential consequences of a major nuclear accident are extremely far-reaching in terms of both time and geography. Land requiring decontamination may be located over 250 miles^{45/} from the accident site, while the area requiring some decontamination measures may approach 3200 square miles.^{46/} Thus, decisions regarding nuclear power plant safety are global in nature. Policy makers must recognize the interstate and international implications of their decisions regarding LWR power plant safety.

The WASH-1400 consequence model includes two strategies for reducing the long-term health consequences of radioactivity deposited on land or structures following a major accidental release. These are (a) interdiction (denial of the use of land for a period of time either by relocating people or by impounding milk and crops) and (b) decontamination of land and property. The radiation dose criteria on which the level of implementation of these measures depends has a significant impact upon the extent of subsequent health effects and the costs of mitigating them.

Two significant questions here are: How much long-term external radiation dose and internal radioactivity are members of the public to be allowed upon returning to a contaminated area, and how effectively and at what cost can reactor-released radioactivity be removed from land and buildings? WASH-1400 answers the first by reference to ten-year old guidelines of the former Federal Radiation Council (FRC), to recommendations of the International Commission on Radiological Protection, and to those of the Medical Research Council (MRC) of Great Britain. The WASH-1400 authors chose to adapt from the FRC and MRC models. (See WASH-1400, Appendix VI, Table VI 11-6). The resulting criteria, when taken together, suggest that those living in rural areas will be allowed 10 rem of whole body dose in a 30-year period, while those living in urban areas will be allowed 25 rem, in the same period of time. Although

the doses may be unavoidable following a major accident, they are nonetheless in contrast to the exposure standards recently promulgated by the U.S. EPA for normal operation of the entire nuclear fuel cycle. (Federal Register, 42, No. 9, pp. 2558-2561, January 13, 1977: .0.025 rem per year to the whole body including bone marrow, and 0.075 rem per year to the thyroid which are equivalent to 0.75 rem and 2.25 rem, over 30 years, respectively.) Over a 30-year period, the WASH-1400 criteria are 13 to 33 times less stringent for whole body than the U.S. EPA's. Thus, choosing the appropriate threshold level for interdiction or decontamination may be a difficult policy decision. Therefore, research to enable federal authorities to develop such criteria for the areas surrounding existing and planned LWR power plants should be included in the LWR safety research program.

One of the largest potential problems following a major nuclear accident may well be sorting out the process required to make the decisions necessary for prompt recovery from an accident, particularly if radioactive contamination crosses state or national boundaries. In some ways, the consequences of a major nuclear accident could be similar to the "acid rain" currently plaguing the east coast of the United States. The acid rain is causing environmental damage but no blame can be established since there is no way of determining the

exact source of the sulfur releases. Fallout resulting from a major nuclear accident could have similar far-reaching damaging implications with one major difference. There would be no question as to the source of material released subsequent to a major accident.

Even if the government of the United States decides that the land contamination standards assumed in WASH-1400 are adequate for the protection of U.S. citizens, there is no assurance that Mexico or Canada will consider these same standards acceptable, particularly if the source of the contamination is from outside their national boundaries. The delays that would result from international negotiations following such an accident would certainly add to the risk and consequences of the event.

We recommend that criteria be developed and accepted by neighboring countries and states within the U.S. so that expeditious recovery from a nuclear accident could be accomplished with a minimum of bureaucratic delay. In order to accomplish this, nationally and internationally accepted land contamination criteria needs to be developed, plus international standards on general population allowable dosage are a necessity. Research to enable federal authorities to establish such criteria should be included in the LWR safety research program.

3.5.1: Decontamination Methods and Potential Costs

WASH-1400 examined the consequences of lower or higher dose criteria for land decontamination in a series of parametric studies (Appendix VI, Section 13.3). Although WASH-1400 presents no graphical results below a dose of 5 rem in 30 years, the trend of their results at that cut-off shows an exponential growth both in the land area that will require long-term interdiction, and in the area which requires decontamination for re-inhabitation. Areas delivering doses lower than 5 rem in 30 years may still require costly decontamination. The dollar costs of reactor accidents may, therefore, be significantly underestimated in WASH-1400.

Missing, too, from WASH-1400 are answers to the unresolved questions as to whether it would be technically and economically feasible to decontaminate land, buildings, and pavements exposed to releases of radioactivity in a reactor accident. The WASH-1400 Appendix VI discussion of this question concedes that experimental evidence is not adequate to support any assumptions on the effectiveness of wet decontamination, because data in this field were generated, for the most part, for the planning for reclamation in the event of nuclear war (when chemical and physical characteristics of the contamination are significantly different).^{47/} Hence, only total removal of contaminated hard surfaces is postulated, while contaminated land is either removed and buried or deep plowed. The latter method does not solve the problem of root uptake of radioactivity into vegetation. WASH-1400 argues that on the average, over large areas,

95% of surface contamination can be removed economically;^{48/} but without more extensive experience, it is questionable that this is so. The WASH-1400 authors qualify their estimate with words such as "appears,"^{49/} "considered,"^{50/} and "not directly applicable."^{51/}

We recommend that the LWR safety program should include research to evaluate technically and economically the appropriate techniques for decontamination following a reactor accident. Such studies should be completed on a high priority.

3.6: EMERGENCY RESPONSE PLANNING

State and local authorities are responsible for the adequacy of nuclear plant emergency plans. Implementation of a well planned emergency response plan could significantly reduce the prompt health effects resulting from an accidental radioactive release from a nuclear plant. Evacuation is the "last ditch" action that must be taken when the design adequacy of the reactor equipment fails to protect the public.

The responses to the accident at TMI-2 once again teaches that the current planning efforts are not adequate. In theory, response plans look good; but in practice, they do not work. For example, one year ago, an NRC Licensing Board found that the emergency response plans for Three Mile Island were perfectly adequate. After the accident, the senior federal official in charge

of federal disaster assistance found that the pre-emergency planning and readiness was totally inadequate.^{52/}

Recent government studies also point up the inadequacy of the current emergency response planning approach and recommend changes in that approach. Specifically, in December, 1978, EPA and the NRC published a joint report to guide federal, state and local governments in formulating emergency response plans around nuclear power plants.^{53/} In summary, the report concludes:

- A spectrum of accidents (not the source term from a single accident sequence) should be considered in developing a basis for emergency planning.
- The establishment of Emergency Planning Zones of about 10 miles for the plume exposure pathway and about 50 miles for the ingestion pathway is sufficient to scope the areas in which planning for the initiation of predetermined protective action is warranted for any given nuclear power plant.
- The establishment of time frames and radiological characteristics of releases provides supporting information for planning and preparedness. ^{54/}

Additionally, on March 30, 1979, the General Accounting Office published a report entitled "Areas Around Nuclear Facilities Should Be Prepared For Radiological Emergencies." That report made the following recommendations to the NRC:

- Allow nuclear power plants to begin operation only where state and local emergency response

plans contain all the Commission's essential planning elements. In addition, the Commission should require license applicants to make agreements with state and local agencies assuring their full participation in annual emergency drills over the life of the facility.

- Establish an emergency planning zone of about 10 miles around all nuclear power plants as recommended by the Environmental Protection Agency/Nuclear Regulatory Commission task force, and require licensees to modify their emergency plans accordingly. 56/

Other reports assessing emergency response planning after the TMI-2 accident, such as the report prepared for the Governor of California (included herein as Appendix E), proposed recommendations designed to provide greater protection to the public in the event of an LWR plant accident.

In addition, consideration must also be given to the effectiveness of the decision process by which an evacuation order is initiated. The quality of the information on which such a decision will be based must also be evaluated. Past experience at sites where major problems have occurred give indication of the difficulty of timely and accurate decisions under stress conditions. Faulty instrument indications, chaotic control room conditions, and above all, the uncertainty of what really was going on inside the "blind" containment was a common factor at the Brown's Ferry Fire, at the Dresden blowdowns, and at the Three Mile Island accident. Effective evacuation depends

firstly upon a timely decision to initiate the plan prior to confirmation that a core meltdown really exists.

We recommend that the following aspects of emergency planning be evaluated in the LWR safety research program:

- a. Evacuation planning should be based on the maximum accidents consequences, not the present design basis accident.
- b. Evacuation initiation should not rely on a potentially subjective and slow human process. An automated alarming and initiating system should be developed and implemented.
- c. Health facilities for treating irradiated members of the public should be identified.
- d. Population growth and siting of other facilities around nuclear plant sites should be controlled.
- e. Public training programs, including possible "mock" drills, should be instituted.
- f. The use of potassium iodide tablets, a mild laxative, and a filter mask should be considered as ways to diminish doses from internal emitters in the first day of an accident.

3.7: RADIATION EXPOSURE OF PLANT EMPLOYEES

Compilation of occupational radiation exposure reports from operating reactors has shown that annual exposures to station and contractor personnel has generally been increasing with time (see Table 3-3). In response, the NRC has initiated Task Action Plan B-34. Task B-34 is intended to provide an improved basis for reviewing designs and LWR operations to assure that occupational radiation exposure is maintained as low as is reasonably achievable (ALARA). At the present time, occupational exposures reported from operating nuclear facilities are averaging nearly 600 man-rem per reactor year, while the expected value for the annual accident exposure to the public associated with the plants analyzed in the WASH-1400 is predicted to be approximately 250 man-rem per reactor year. Thus, the occupational exposure exceeds the predicted commitment associated with accidents at the present time. While a meaningful comparison of the risk associated with occupational exposure to the public risk associated with accidents is difficult, it should be recognized that the level of occupational exposures noted are actually being received. Thus, the occupational exposures are potentially of high risk significance; reduction of these exposures can be important in reducing the overall radiologically-associated risks associated with the nuclear reactor industry.^{57/}

TABLE 3-3

SUMMARY OF ANNUAL EXPOSURES
REPORTED BY NUCLEAR POWER FACILITIES

1973-1977 58/

Year	Reactor Type	Number of Reactors Operating Full Year	Total Number of Man-Rems	No. of Workers With Measurable Exposure	Total Megawatt-Yrs. Generated	Average Annual Dose (Rems/Person)	Average No. of Man-Rems Per Reactor	Average No. of Workers Per Reactor	Man-Rems Per Megawatt-Yr.
1973	PWR	12	9,399	9,440	4,065	1.00	783	787	2.3
	BWR	12	4,564	5,340	3,344	0.85	380	445	1.4
	Total	24	13,963	14,780	7,409	0.94	582	616	1.9
1974	PWR	20	6,627	9,697	6,821	0.68	331	485	1.0
	BWR	14	7,095	8,769	4,042	0.81	507	626	1.8
	Total	34	13,722	18,465	10,863	0.74	404	543	1.3
1975	PWR	26	8,268	10,884	11,983	0.76	318	419	0.7
	BWR	18	12,611	14,607	5,786	0.86	701	812	2.2
	Total	44	20,879	25,491	17,769	0.82	475	579	1.2
1976	PWR	30	13,807	17,588	13,251	0.79	460	586	1.0
	BWR	23	12,626	17,859	8,312	0.71	549	776	1.5
	Total	53	26,433	35,447	21,563	0.75	499	669	1.2
1977	PWR	34	13,469	20,878	16,431	0.65	396	614	0.8
	BWR	23	19,042	21,338	9,133	0.89	828	930	2.1
	Total	57	32,511	42,266	25,584	0.77	570	742	1.3

*The figures on this table are based on the number of nuclear power reactors that had been in commercial operation for at least one year as of December 31 of each of the years indicated.

The problem of maintaining occupational radiation exposure in compliance with ALARA requirements is just beginning to be addressed seriously by the industry for new designs; yet the problem is particularly acute for older operating plants, particularly for BWR's with vessel and pipe cracks and PWR's requiring steam generator replacement. A presentation by the NRC Staff to the ACRS on October 6, 1978 ^{59/} reveals that the NRC was in the process of upgrading occupational radiation exposure from a Category B ^{60/} generic problem to a problem of highest priority. Decommissioning of reactor plants at end of life, Generic Task B-64, was also under active consideration for upgrading. However, as indicated in NUREG-0510, these tasks were not eventually included in the list of 17 "unresolved safety issues" reported to Congress in January, 1979. These changes were being considered in view of the current request before the NRC Commissioners to substantially reduce allowable occupational exposure levels, maybe by as much as a factor of 10. Such reductions will undoubtedly occur, if not immediately, within the relatively near future, and it is essential that all nuclear plants be thoroughly reviewed for compliance with ALARA guidelines as early as possible. This should preferably be done at the construction permit review, but a thorough review at the operating permit stage

is vastly superior to attempting to make corrections after the plant becomes radioactive.

In summary, the operation, maintenance, and retrofit of nuclear plants is requiring ever increasing exposure of plant personnel due to the increasing levels of radiation within the plants, and the increase in plant numbers and size. Regulatory Guide 8.10, Operating Philosophy for Maintaining Radiation Exposure as Low as Reasonably Achievable has been issued, but as the title implies, it is philosophical in nature and strict implementation of it has not been required. The NRC should initiate a comprehensive LWR safety research program to evaluate all aspects of this problem, and to establish requirements to ensure that the future health impact is minimized. The NRC should include consideration of such things as plant configuration, material selection, shielding, automated tools, plant decontamination and decommissioning, and the acceptability of current exposure limits. The impact on plant operability, if individual exposure limits are reduced, should also be evaluated by the NRC in the safety research program.

SECTION 3

REFERENCES

- 1/ NUREG/CR-0400, Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission, U.S. Nuclear Regulatory Commission, Washington, D.C., September, 1978, p. vii.
- 2/ Jan Beyea, A Study of Some of the Consequences of Hypothetical Reactor Accidents at Barseback, Swedish Energy Commission Report DsI 1978:5, January, 1978, p. I-1.
- 3/ WASH-1400, Appendix VI, p. 2-5.
- 4/ WASH-1400, Appendix VI, p. 1-3.
- 5/ SECY-78-137, Memorandum to the NRC Commissioners on Assessments of Relative Differences in Class 9 Accident Risks in Evaluations of Alternatives to Sites With High Population Densities, March 7, 1978.
- 6/ NUREG-0371, Task Action Plans for Generic Activities - Category A, U.S. Nuclear Regulatory Commission, Washington, D.C., November, 1978, p. A-33/1 & A-33/2.
- 7/ Docket No. 50-272, Salem Unit 1, "NRC Staff Response to Board Question No. 4 Regarding the Occurrence of a Class 9 Accident at TMI," by B.H. Smith, August 24, 1979
- 8/ Report to the American Physical Society by the Study Group on Light-Water Reactor Safety, Rev. Mod. Phys. 47, S1, 1975.
- 9/ Ibid 2, p. I-4.
- 10/ Reactor Safety Study (WASH-1400): A Review of the Final Report, U.S. Environmental Protection Agency, Office of Radiation Programs, EPA-520/3-76-009, p. 2-14.
- 11/ H. Kendall, et al, The Risks of Nuclear Power Reactors: A Review of the Reactor Safety Study WASH-1400 (NUREG-75/014), Union of Concerned Scientists, Cambridge, Massachusetts, August, 1977, page 67.
- 12/ Ibid 2, p. II-8.

- 13/ Neal S. Nelson, "Review of the EPA Position on WASH-1400, the Reactor Safety Study Health Effects Estimates," given on August 25, 1977 meeting of Risk Assessment Review Group.
- 14/ Ibid 10.
- 15/ EPA and NRC correspondence, letters dated July 18, 1977; April 15, 1977; April 5, 1977; December 20, 1976; November 18, 1976; and July 2, 1976 (two letters).
- 16/ Nuclear Energy Policy Study Group, Nuclear Power Issues and Choices, The Ford Foundation, 1977.
- 17/ Department of the Interior, Comments of February 9, 1977, on Reactor Safety Study on Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants (WASH-1400).
- 18/ Frank von Hippel, "Notes for Discussion with ACRS Working Group on NRC's Reactor Safety Study," January 4, 1977.
- 19/ Frank von Hippel, "Looking Back on the Rasmussen Report," Bulletin of Atomic Scientists, February, 1977.
- 20/ Joel Yellin, Testimony before the Working Group on the Reactor Safety Study, ACRS, January 4, 1977.
- 21/ Joel Yellin and NRC correspondence. December, 1975 to December, 1976; Yellin to Levine, December 15, 1975; Yellin to Mason, March 3, 1976; Mason to Yellin, May 14, 1976; Yellin to Mason, July 1, 1976; Mason to Yellin, July 26, 1976; Yellin to Mason, September 10, 1976; Levine to Yellin, October 18, 1976; Yellin to Levine, October 26, 1976; Levine to Yellin, December 3, 1976. Includes Joel Yellin; "The Nuclear Regulatory Commission's Reactor Safety Study," The Bell Journal of Economics, 7, (1), Spring, 1976, pp. 317-339; Letter from Thomas Neff to Richard Wilson dated June 8, 1976.
- 22/ "Observations on the Reactor Safety Study," Report prepared by the Subcommittee on Energy and the Environment of the Committee on Interior and Insular Affairs, January, 1977.
- 23/ Minutes of Meeting One of the RARG (8/24 & 8/25, 1977), issued by John H. Austin, U.S. Nuclear Regulatory Commission, Washington, D.C., April 18, 1978, pp. 16 to 21.
- 24/ Ibid 23, Attachment D, p. 12 to p. 20.
- 25/ Ibid 23, Attachment D, p. 16.

- 26/ Minutes of Meeting Eight of the RARG (4/3 to 4/4, 1978), issued by John H. Austin, U.S. Nuclear Regulatory Commission, Washington, D.C., February 14, 1979, p. 12.
- 27/ Ibid 26, p. 16.
- 28/ Ibid 26, Attachment G.
- 29/ J.L. Sprung and H.W. Church, "Effects of Wind Shear on the Consequence Model of the Reactor Safety Study," Sandia Laboratories Report No. SAND 76-0619, January, 1977.
- 30/ J.L. Sprung and H.W. Church, "Sensitivity of the Reactor Safety Study Consequence Model to Mixing Heights," Sandia Laboratories Report No. SAND 76-0618, January, 1977.
- 31/ L.T. Ritchie, W.D. Brown, J.R. Wayland, "Effects of Rainstorms and Runoff on Consequences of Nuclear Reactor Accidents," Sandia Laboratories Report No. SAND 76-0429, October, 1976.
- 32/ J.R. Wayland, "Conditional Probability of Intense Rainfall Producing High Ground Concentrations from Radioactive Plumes," Sandia Laboratories Report No. SAND 76-0746, March, 1977.
- 33/ A.J. Russo, "Reactor Accident Plume Risk Calculations," Sandia Laboratories Report No. SAND 76-0340, July, 1976.
- 34/ A.J. Russo, J.R. Wayland and L.T. Ritchie, "Influence of Plume Rise on the Consequences of Radioactive Material Releases," Sandia Laboratories Report No. SAND 76-0534.
- 35/ Ibid 26, Attachment I.
- 36/ Comments by Dr. Robert Erdmann, Science Applications Inc., Minutes of Meeting Five of the RARG (1/5 to 1/6, 1978), issued by John H. Austin, U.S. Nuclear Regulatory Commission, Washington, D.C., June 27, 1978, Attachment D, p. 57.
- 37/ Comments by Dr. Joseph Murphy, U.S. Nuclear Regulatory Commission, Minutes of Meeting Two of the RARG (10/5 & 10/6, 1978), issued by John H. Austin, U.S. Nuclear Regulatory Commission, Washington, D.C. May 9, 1978, p. 16.
- 38/ Ibid 8, p. 109.
- 39/ WASH-1400, Appendix VII, p. 47.
- 40/ Ibid 11, p. 96.

- 41/ WASH-1400, Main Report, p. 76.
- 42/ WASH-1400, Main Report, p. 134.
- 43/ NUREG-0603, Comments on the NRC Safety Research Program Budget, ACRS Report, U.S. Nuclear Regulatory Commission, Washington, D.C., July, 1979, p. 3-12.
- 44/ Ibid 43, p. 3-13, 3-14, and 3-15.
- 45/ Ibid 2, p. I-12.
- 46/ WASH-1400, Appendix VI, p. 13-47.
- 47/ Ibid 46, p. 11-19.
- 48/ Ibid 46, p. 11-20.
- 49/ Ibid 46, p. K-12.
- 50/ Ibid 46, p. 11-20.
- 51/ Ibid 46, p. 11-19.
- 52/ Statement by William H. Wilcox, Administrator, Federal Disaster Assurance Administration, before the President's Commission on the Accident at Three Mile Island.
- 53/ NUREG-0396, EPA 520/1-78-016, Planning Basis for Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Washington, D.C., December, 1978.
- 54/ Ibid 53, p. 24.
- 55/ EMD-78-110, Areas Around Nuclear Plants Should be Better Prepared for Radiological Emergencies, U.S. General Accounting Office, Washington, D.C., March 30, 1979.
- 56/ Ibid 55, p. V.
- 57/ M. Cunningham, J. Murphy, M. Taylor, Preliminary Draft - Summary Report on a Risk Based Categorization of NRC Technical and Generic Issues, U.S. Nuclear Regulatory Commission, Washington, D.C., p. I-22 to I-24.
- 58/ NUREG-0463, Occupational Radiation Exposure - Tenth Annual Report - 1977, U.S. Nuclear Regulatory Commission, Washington, D.C., October, 1978, p. 10.

59/ ACRS Full Committee Meeting Transcript, October 6, 1978.

60/ NUREG-0471, Generic Task Problem Descriptions - Category B, C, and D Tasks, U.S. Nuclear Regulatory Commission, Washington, D.C., June, 1978.

SECTION 4

SAFETY ASSURANCE EFFECTIVENESS

4.1: INTRODUCTION

Safety Assurance is an unofficial descriptive title used in this report to encompass the essential combined structure of safety-related philosophical, developmental concepts, administrative, and procedural methods and processes which must be effectively performed to assure that the total nuclear establishment actually produces facilities that meet the desired safety goals. Quality assurance and the issue of regulatory effectiveness are two well known parts of this structure which have received much attention and discussion, but which continue to elude acceptable realization. There are many additional aspects of this safety assurance structure, any one of which can (and does) break down, resulting in degradation of the safety actually achieved. The breakdown can occur throughout a wide range of the lengthy process required to produce and operate a nuclear plant, ranging from a failure of the basic design criteria, to lack of quality in the construction process, and on to the relatively expected, but unpredictable factor of human errors committed under abnormal or emergency operating conditions.

To further complicate the matter is a complex array of regulations, organizations, interfaces, and divided responsibilities that is continuously changing throughout the life of

the nuclear project as it moves from the conceptual stage, into design and construction, and finally into operating status. Not only does the nature of the project and its needs change, the organizations and personnel within a given institution change from phase to phase. There are four basic phases to consider:

1. Design
2. Licensing
3. Construction
4. Operation

Within the three major entities involved, the vendors, the utility, and the regulatory agency, organizations are generally emplaced to implement the specific activities of each of these phases. For example, the typical nuclear steam supply system vendor will have a separate design group, a licensing operation, a project (construction) organization, and finally an operating liaison group. Each one of these groups deals with a (similar) counterpart in the utility or the NRC. Voluminous communications, procedures, and documentation are required to establish and define the various interfaces. With the divided responsibilities, periodic transfers of management, and the changing emphasis for the project needs, there are many opportunities for errors, or less than optimum attention, and some functions are overlooked or are not completed. Last but not least, the difficulty of managing all aspects of this complex situation make time the enemy.

Difficult confirmatory programs or regulatory decisions are delayed, but the concrete continues to pour per the construction schedule. A prime example of this process of indecision is the near 10-year consideration of what to require for the resolution of ATWS, or the LOFT program that began more than 13 years ago but is incomplete at this late date. The technical complexities of such issues, coupled with the "impenetrable shield of middle management" present not only in the NRC, as discovered by the Kemeny Commission, but prevalent throughout an industry made up primarily of large organizations, acts to make timely resolution of such problems near impossible.

All of these important issues might be considered to be theoretically covered by the quality assurance programs required by the NRC's 10 CFR 50 Appendix B. The 18 QA criteria contained in Appendix B basically cover all of the safety assurance requirements, and if the 18 criteria were successfully carried out in a timely manner, they would probably provide satisfactory implementation of the safety program. It would certainly be true that the quality assurance program specified in Appendix B, aggressively enforced by a knowledgeable agency with adequate regulatory powers, should result in the achievement of substantial improvement of light water reactor safety.

In the course of our review for this Study, as well as in our ongoing review of light water reactor safety programs, we have encountered many suggestions and have developed numerous

recommendations on how the safety assurance structure could be improved. Most of these improvements are not new and have been documented on numerous occasions in past reviews and studies. They are difficult to achieve for the various reasons of technical and institutional complexity expressed previously, but they are well worth continued effort and study if any significant safety improvement is to be achieved. For purposes of organization, we have consolidated the broad assortment of issues into six general categories. The six categories which are addressed in some detail in subsequent portions of this section are:

1. Design Process Effectiveness.
2. Quality Assurance Implementation.
3. Regulatory Enforcement.
4. Person-Machine Interface.
5. Licensing Process.
6. Other

4.2: DESIGN PROCESS EFFECTIVENESS

The design process is the step required to translate the scientific concept into a set of specifications, drawings, and instructions suitable for procurement and construction of the project. Design process effectiveness is a measure of the degrees to which the system and components work as the designer intended. There are a number of important issues in this design process which are either deficient, or need substantial

improvement in order to make the system work adequately.

These issues are:

4.2.1: Assignment of Responsibility

Assignment of responsibility is critical to ensure that proper actions are taken by the technical designers to complete the design process and follow the performance of the equipment into the operating phase and beyond. Trouble is often encountered in this area in the post-operational phase of the project after the original designer of the system or equipment no longer has contractual responsibility. A prime example of this situation was observed at Three Mile Island where individuals in the vendor organization, in the NRC, and in another utility identified a design weakness or deficiency but because of the uncertainty of responsibility, could not overcome the inertia of the system so as to communicate the action that should be taken. Appendix B Criterion III indicates that design changes shall be subject to design control measures commensurate with those applied to the original design and are to be approved by the organization that performed the original design. Unfortunately, this is difficult to accomplish in a commercial endeavor where responsibilities are governed by contracts that generally terminate at time of completion of start-up tests. A method of maintaining active participation of the designer throughout the operating life of the project is needed.

4.2.2: Design Criteria Adequacy

The General Design Criteria contained in Appendix A of 10 CFR 50 are in need of a thorough re-evaluation, both as to content and as to interpretation. The Three Mile Island accident demonstrated clearly the inadequacies of the "single-failure criterion" and the ongoing reviews will undoubtedly define other areas of inadequacy. In addition, the methods by which the General Design Criteria are implemented by the designers need thorough study. A large portion of the nuclear plant system consists of relatively standard hardware and equipment which receives "paper compliance" through narrow interpretation of the criteria. A thorough review of the General Design Criteria, and of all levels of the procedures for their implementation should be performed.

4.2.3: Codes and Standards

Much of the nuclear system equipment and hardware is designed to be in compliance with various codes and standards developed by various organizations associated with the industry. These standards, for the most part, are developed by a consensus of committees made up of persons working within the nuclear industry and as such, often serve to protect the interests of the industry rather than the interests of the public health and safety. In addition, standards development is an extremely slow process, taking years for development and even a longer period

of time before actual hardware changes appear in the field. Deficiencies of the standards and certification procedures were addressed in a hearing before the Federal Trade Commission and testimony presented before that body by Richard B. Hubbard and Gregory C. Minor on May 18, 1979 is recommended reading for further details on those deficiencies.^{1/} Changes to the standards development system should include requirements of adequate technical bases, funding of standards representatives from the public, more formal control of standards committee membership, and of the review process, and timeliness of issuance. Unless such changes are made, the standards program will continue to provide the illusion that all technical bases are well-founded.

4.2.4: Design Verification

Design verification is a step in design control required by Criterion III of 10 CFR 50 Appendix B. Design verification may be accomplished by a number of alternate methods such as design reviews, alternate or simplified calculational methods, or by the performance of suitable testing. In the past, too much of the design verification has been a paper work formality performed primarily "after the fact" by means of relatively unsophisticated analytical methods. Critical components and systems should be identified and a greater reliance placed on confirmatory testing before hardware commitments are made.

4.2.5: Confirmatory Testing

Confirmatory testing, identified above in design verification, is often used to verify satisfactory performance of critical components and equipment. The long delays suffered by the LOFT program, the untimely performance of the Mark I and Mark II containment test programs, and the TMI-identified need for comprehensive testing of power-operated relief valves gives strong evidence for the inadequate handling of this critical design step; substantial improvement is needed.

4.2.6: Equipment Qualification

Included among the essentials for safe design is the requirement that equipment important to safety be designed to withstand the effects of the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss of coolant accidents. Such qualification testing is required by IEEE Standard 323-1974, but experience has shown that this is not being adequately done. The instrument failures following the Three Mile Island accident strongly illustrate both the necessity and deficiencies of this step. Of particular concern is the need to demonstrate that the effect of aging of equipment has been adequately considered. A substantial effort is required to develop qualification methods and implement modifications indicated by the results of such tests.

4.2.7: Experience Feedback and Analysis

A more effective system of experience feedback and analysis is needed to ensure the required improvements to the design process are made. Some information is now becoming available through the Nuclear Plant Reliability Data System. This system should be expanded in scope, and be supplemented by a requirement that the original designers perform a timely review and disposition of field failure reports. Notification of users of similar equipment should be required, rather than at the vendor's discretion.

4.2.8: Design Process Conclusions

- These design process deficiencies and recommendations are not easily translated into a detailed and achievable research and development program. It is absolutely essential, however, that improvements in the design process be achieved if real improvement is to be obtained in the operating plant.

4.3: QUALITY ASSURANCE IMPLEMENTATION - GENERAL CONCERNS

The Quality Assurance program mandated for commercial nuclear power programs by 10 CFR 50 Appendix B is intended to assure that the safety and reliability features specified by the regulations and the system designers are in fact implemented by the many different organizations involved. Successful implementation of the QA program is essential to achieving expected levels of safety. There is substantial evidence that an adequate implementation of QA programs is not presently being attained. The following are of general concern:

4.3.1: QA - I & E Program Effectiveness

The NRC's Inspection and Enforcement (I&E) program is intended to provide an independent verification that the nuclear facilities' structures, systems, and components are designed, manufactured, installed, and operated in strict accordance with the applicable quality assurance requirements. In the past, the I&E program has not fulfilled this intended function.

The "after the fact" discovery by the NRC of quality deficiencies at the North Anna,^{2/} Browns Ferry,^{3/} and the Davis Besse plants raises serious questions about the adequacy of the whole NRC Inspection and Enforcement Program. In particular, questions need to be answered about the NRC policy of relying on builders for primary inspections with NRC officials serving as only auditors.

In partial response to the numerous criticisms of the NRC I&E practices, in May, 1976, the NRC provided Sandia Laboratories of Albuquerque, New Mexico, with over a quarter of a million dollars of funding to conduct a comprehensive, independent assessment of the NRC activities related to the review, approval, and inspection of quality assurance programs at commercial nuclear power plants. Specifically, the study assessed the effectiveness of the overall philosophy of the NRC QA program and the relative strengths and weaknesses of the practices employed to assure a high standard of quality assurance for nuclear reactors.^{4/}

The Sandia Study's final report was released as a NUREG series document in September, 1977^{5/} While the 16 recommendations of the study group were carefully worded in a positive manner so as to not imply that the existing NRC I&E program is inadequate, the message is still clear. The report states in the summary that "based on the results of our survey and the stringent demands for reactor safety, we conclude that further improvements are warranted in both industry quality assurance programs and NRC regulations of these programs." Specifically, the study concludes that "routine direct NRC inspection and testing of hardware be increased, and that data pertinent to quality decisions made in the construction and operation of a plant be evaluated by the NRC on a routine basis. This includes the evaluation, for example, of radiographic and ultrasonic test data."

In 1978, a study conducted by the General Accounting Office (GAO) described the following weaknesses in the NRC's I&E program during nuclear power plant construction:

"Although the Nuclear Regulatory Commission is responsible for assuring that nuclear power plants are constructed safely, it has not been independently testing the quality of construction work. The Commission should do this, plus

- improve its inspection and reporting practices,
- use the inspectors' time and talents more efficiently, and
- better document its inspection findings.

The Commission is aware of the need for improvements and has made some changes, one of which is the assignment of resident inspectors to selected reactors under construction."^{6/}

Other studies conducted by and for the NRC have identified opportunities for improvement in the I&E program.^{7/} Since a key factor in assessing the potential risk of a nuclear plant is the assumption of a disciplined, thorough quality assurance program, the inadequacies in the NRC's I&E oversight of the plant owner's quality assurance program may allow deficiencies in the program implementation which will pose a significant hazard to the public health and safety. These deficiencies should be corrected.

4.3.2: QA - Lack of Adequate As-Built Drawings

The Browns Ferry Fire and the TMI-2 accident highlighted a weakness in the quality assurance programs related to the availability of "as-built" drawings. At Browns Ferry, no one knew which or when a vital service would be lost as the fire progressed. The operators, based on the drawings available, did not know the physical locations of the various circuits.^{8/} Likewise, one of the lessons learned from the TMI-2 accident was that neither the power plant owner nor the NRC had a set of "as-built" drawings readily available. The TMI-2 Lessons Learned Task Force concluded that, ".....as-built drawings reflecting the actual configuration of the plant were either not available or contained erroneous information. This situation contributed to delays in accident recovery."^{9/} As-built drawings should be current and available.

4.3.3: QUALITY ASSURANCE CONCLUSIONS

The impact of such deficiencies on safety is significant, and the misinformation that is a result of this uncertainty is equally of concern. For example, the WASH-1400 methodology depends on the existing data and certain assumptions regarding quality assurance. The Browns Ferry Fire revealed some checks on QA were not sufficient and thus all calculations of risk could be invalidated. In order to ascertain the usefulness of the risk assessment methodologies, it is necessary to address the effectiveness of QA. There are three issues: the ramifications of any degradations in QA efforts over the years; how to include any such decline in risk assessments; and the impact of an insufficient QA program on risk assessment.

- It is recommended that these general QA issues be specifically addressed by the ongoing R&D program. Corrective action is needed to resolve the QA deficiencies, and quantification of their impacts on risk assessment is essential to making future priority decisions.

4.4: REGULATORY ENFORCEMENT

Regulatory enforcement is that area of the safety assurance structure that provides the NRC with the authority and intelligence to ensure that the nuclear regulations are being followed in a timely and effective manner. Numerous audits and reviews identified in Section 4.3 of this report indicate that

substantial improvement of regulatory enforcement effectiveness is both needed and possible. Over the past several years, a significant amount of criticism has been generated on these subjects and the NRC has responded with some improvements. The following items are those which are judged to be in need of additional effort:

4.4.1: Audits and Inspections

Audits and Inspections play a critical role in the NRC's determination of the licensee and his suppliers compliance with the applicable regulations. The main criticisms of the NRC's audit and inspection program have to do with the infrequency of inspections, the shallowness of the audit, and the predictability of not only when the audit will occur, but what specific questions will be asked. Changes should be made in the system of audits and inspections to strengthen this function in these areas of weakness.

4.4.2: Supplier Inspections

Supplier inspections are a part of the overall inspection and enforcement program in which the NRC periodically inspects some of the licensee's suppliers to determine compliance with applicable regulations. In practice, the NRC's inspection of suppliers is generally limited to a review of quality assurance plans and procedures. The inspection seldom, if ever, evaluates the quality condition of the hardware or components being produced. A more aggressive program of supplier inspections should be implemented.

4.4.3: Resident Inspectors

Resident inspectors are beginning to be utilized by the NRC at the more active construction sites and at operating plant sites. This is a program that has been strengthened somewhat following the Three Mile Island accident, but it has been in place such a short period of time that no real determination can be made of its effectiveness. This program needs to be carefully monitored and expanded with a particular emphasis being placed upon a reporting format in information flow so as to make the best use of the resource available. Additionally, resident inspectors should be emplaced by the NRC at the licensee's major suppliers such as the reactor system vendor and the architect engineer's facility.

4.4.4: Regulatory "Muscle"

Regulatory "muscle" is a term used to indicate the system of fines and/or penalties available to the regulator to assist in achieving attention or action when significant deficiencies are discovered. Even intentional cases of non-compliance (of 10 CFR 21.21) are limited to civil penalties not in excess of \$5,000 for each failure to provide notice, and are not to exceed \$25,000 within any period of 30 consecutive days. Such relatively small civil penalties are insignificant when compared to the several hundred thousand dollars per day fuel differential costs involved with a typical nuclear power plant. The option open to the NRC

is, of course, to issue a shutdown order or terminate the license. This would in general provide more of a penalty to the licensee, but it is not clear whether such penalty would accrue to the licensee or simply be passed on as a hardship or expense to the licensee's ratepayers. A system of significant penalties which would directly accrue to the licensee's management or stockholder levels should be developed and aggressively administered to ensure adequate regulatory enforcement.

4.4.5: Dissenting Views

Dissenting views are theoretically solicited by the NRC. Information is presented in 10 CFR 21.2 to provide a mechanism for the reporting of deficiencies, including provisions for anonymity. Experience with individuals stepping forward to present their concerns has generally shown that the regulatory agency meets such claims with intense skepticism. An unreasonable burden of proof is expected to be borne by the dissenter. It is not reasonable, of course, for the NRC to terminate operations or halt construction at the mere expression of concern by an individual, but substantially more objectivity should be shown in attempting to get to the root of the dissenter's technical concerns.

4.4.6: Inspecting the Inspector

Periodic audits of the NRC's basic inspection and enforcement program are desirable to ensure that the regulator is

performing his responsibilities adequately. In the past, this has been done ad hoc by the GAO and by other organizations and the NRC has arranged to have audits performed by outside organizations. The performance of a mandatory audit should be made a requirement that is to be performed approximately annually (or more frequently if the need arises). Such audits should be reviewed publicly and corrective action programs developed in response to the deficiencies identified.

4.4.7: Regulatory Conclusions

- It is recommended a safety improvement study be initiated to determine how the preceding recommendations concerning regulatory enforcement could be adopted. This would serve to make the NRC a much more efficient organization which could in turn improve the safety of LWR operation. This ultimately would shorten the licensing time and improve plant performance. This regulatory reorganization and streamlining is a natural follow-on to the Three Mile Island review activities currently underway. The Three Mile Island Inspection and Enforcement Report, NUREG-0600,^{10/} is a good starting point for considering the most important aspects of this subject. "Operational Aspects - Potential Items of Non-Compliance," identified as Appendix I-B of that report contains some 12 pages of potential non-compliance items which presumably could have been avoided by a more effective regulatory enforcement program.

4.5: PERSON-MACHINE INTERFACE

The following group of safety assurance procedures and issues involves those having to do with the minimization of human errors which can occur affecting the safety or physical care of the operating nuclear plant. These items range from the proper utilization of human engineering in the design of the control room, to the measures taken to protect against sabotage or intentional mal-operation of the equipment. The following items are believed to be of significance in the achievement LWR safety:

4.5.1: Human Factors Design

The control room of a nuclear plant is an impressive array of meters, lights, switches and annunciators representing more than a hundred different systems and tens of thousands of device-functions. From this sometimes bewildering array, the operator must make the critical decisions necessary to assess the plant status and assure that action is taken to correct any safety problems which arise. This can place a large demand on the operator in charge of a billion dollar facility which is providing electricity for close to a million people and which could cost the utility company \$500,000 a day for down time.

Historically, the human factors aspect of control room design have been largely neglected in the nuclear industry. A recent study by EPRI summarizes the situation very well:

"Although human factors engineering has a history of over thirty years, the human factors design principles initially developed for military and space programs to ensure operator effectiveness and reliability have not been generally or consistently applied to the design of power plant operational work spaces. For example, four out of the five control rooms reviewed revealed serious violations of anthropometric standards where indications were placed beyond viewing limits and controls were located beyond acceptable reach limits. Control room configurations varied widely, with some designs making for awkward operations necessitating additional manning when compared with more efficient configurations. The extensive use of backrack areas which take operators away from the primary sphere of control and the practice of mirror-imaging control boards in multi-unit control rooms were found to be particularly disadvantageous from the human factors standpoint. Control board designs were generally found to be excessive in size, lacking in functional arrangement of elements (sometimes necessitating a two-man operation of a control and its associated display), and generally lacking in clarity of interrelationships between panel elements. In short, the control boards reviewed had not been designed to promote error-free operation, especially during potentially stressful circumstances"11/

There is a need to set standards for human factors design and implement changes on existing poor designs.

4.5.2: Operator Training and Reactor Simulation

In order to become a reactor operator, the potential candidate must learn reactor principles, the operating processes for the reactor to be operated, and an understanding of how to handle upset conditions. To help with this training process,

most operators are sent to simulator training centers for a combination of textbook instructions and hands-on experience. During the usual simulator training period of less than a week, the prospective operator learns to handle start-up, run, shut-down and upset or abnormal conditions of the reactor. Unfortunately, these simulators are not always identical or, in some cases, even reasonably similar to the plants which will be operated.

The operators at TMI-2 were trained on a simulator at the B&W center which uses a replica of the much smaller and simpler control room from the Rancho Seco plant in Sacramento, California. This may or may not have contributed to their misinterpretation and misoperations at TMI-2; we will probably never know. The existing number of reactor simulators is far below an optimal level. The optimum number and type of simulator facilities should be evaluated and the new facilities installed. There also needs to be more consideration of the degree to which a simulator is representative of the actual reactor to be operated. Finally, the amount of training time must be increased to cover more transient and accident conditions.

4.5.3: Automation vs Manual Operations

Safety of an error-prone manual process may be improved by making changes toward automation or making changes to improve the operator's response. In the wake of TMI, there are

recommendations for both actions. Part of the Lessons Learned Task Force recommendations (see Section 2) call for more reliance on the operator's innovative capabilities, including the operator as an extension of the defense-in-depth concept, and increasing the number of operating procedures to include more transients. However, the TMI experience shows that:

1. The operator's innovative process was not adequate to intuit the vessel water level during the critical part of the accident.
2. The operator, in a few cases, violated one or more of the defenses in depth.
3. The operators were following the wrong procedure but one which actually fit the conditions of the key plant parameters.

The alternative would be to automate as many of these functions as possible. This introduces new problems in equipment reliability and the manual actions including "override" (as in the case of HPI at Three Mile Island). There is no clear-cut safety improvement to be gained by increasing the demands on the operators. On the other hand, automated systems are not the total answer because they too involve maintenance, revision, and calibration and numerous other manual actions subject to human error. There is a need to study the trade-off between manual and automated functions as an approach to safety improvements.

4.5.4: Non-Licensed Operators

Non-licensed operators are those personnel involved with plant operations who are not required by current regulations to be licensed for operation. These personnel who are essential to safe and efficient plant operation, nonetheless, are involved with numerous actions on a day-to-day basis which may affect the safety of the plant through acts of error or omission. Examples of non-licensed operator actions during an emergency situation are in the TMI I&E report. ^{12/} Training of such non-licensed operators is discussed by the utility with the NRC at the time of operating license review but no specific requirement is established in the technical specifications or the regulations. (A recent draft Reg Guide does address the need for radiation protection training of all persons.^{13/}) It is recommended that standards for non-licensed operator training be developed and be made a part of the regulatory process.

4.5.5: Maintenance Personnel and Contractor Training

As is the case with non-licensed operators, substantial numbers of maintenance personnel and sub-contract employees are normally in the plant performing functions that can be critical to plant safety, particularly during accident or emergency conditions. Additionally, maintenance supervisory personnel play critical roles in ensuring that equipment essential to safe operation and shutdown of the plant is not made unavailable through maintenance actions except as permitted by the technical specifications and as concurred in by the licensed operating

personnel. The isolation of both auxiliary feedwater trains which contributed heavily to the Three Mile Island accident is a good example of the maintenance error impact on plant safety. As is the case with non-licensed operators, firm requirements and procedures need to be developed and established for the training and qualification of permanent maintenance personnel as well as for the training of temporary sub-contract personnel that may be involved with safety-related equipment.

4.5.6: Utility Management Training

Utility management training is a subject that has been considered as a likely change to the regulations as a result of the TMI reviews. In most utilities, top-level management are notified almost immediately in the event of abnormal operating conditions and these individuals assume positions of direct responsibility and control in the emergency recovery operations. For example, at Three Mile Island, the Station Manager arrived at the plant site approximately 3 hours after the accident was initiated and declared himself Emergency Director.^{14/} The Station Manager is not licensed, nor is he required to be under the regulations. There is no indication that his action was not correctly taken, but it seems logical that positions of such technical authority should require a licensing action by the NRC. Additionally, upper-level management play a heavy role in making the emergency recovery decisions and should receive formalized

training and certification if they are to act in such capacities.

4.5.7: Sabotage and Intentional Mal-Operation

The question of sabotage and security impact on LWR safety continues to be evaluated; but the quantification of the risk derived from this potential failure continues to remain unquantifiable. The recent intentional damage of nuclear fuel at an east coast nuclear plant by plant employees illustrates the unpredictability of human behavior. The simultaneous blockage of both auxiliary feedwater trains at Three Mile Island, presumably by error, illustrates the susceptibility of nuclear plant equipment to intentional mal-operation. It is recommended that the TMI-type review initiated to evaluate the susceptibility of equipment to unintentional defeating of multiple lines of defense be expanded to include consideration of what changes can be made to strengthen the plant's resistance to intentional actions of this same nature.

4.5.8: Interface Conclusions

- The wide range of human interface issues that can and should be addressed will require the development of a major research and development program to determine how these issues can be satisfactorily dealt with on all plants, both existing and yet to be constructed. This is an area that should receive top priority in the research and development program because it is possible by human action or inaction for the operator to defeat almost any automated system that may be employed to assure

the safety of the plant. The human link, therefore, becomes the potentially weakest link of the safety chain and top priority needs to be given to strengthening it by all appropriate means.

4.6: LICENSING PROCESS

Safety assurance procedures listed under this category have primarily to do with the process covering the review and evaluation of applications for construction permits and operating licenses. The licensing process also extends to the operating phase of the plant as there is a process of on-going technical review required to support the changing conditions of plant condition, operating experience, and basic knowledge of nuclear technology. The following items address those issues of the licensing process that currently are judged to be inadequate or non-existent.

4.6.1: Depth of Review - CP/OL stage

The regulations require that two formal reviews be performed prior to initial operation of a nuclear plant. The first review, the one conducted in conjunction with the construction permit approval process, is to review the proposed details of the plant to determine, among other things, that the plant design will comply with the safety criteria and has reasonable assurance that a safe design will be produced. The second review, conducted at the operating license application stage, theoretically consists

of assuring that the plant was built in accordance with the design proposed at the construction permit stage, that no new information has been developed that would cast doubt on the safety of the plant configuration, and addresses details of utility qualifications to operate the plant, the technical specifications concerning operation, etc. The way in which these reviews are performed, and the time sequence in which they are accomplished, generates two problem areas in terms of providing optimum safety assurance. First, the construction permit review is carried out at an extremely shallow level with regard to actual details of the specific plant design. The primary focus of the review addresses the applicant's commitment to comply with the relevant codes, standards, and design criteria and does very little critical looking at the details of the design. This is perhaps the nature of a complex and lengthy design project, as in many cases there are no details yet produced to actually review. However, this shallow initial review is further compromised at the operating license review by the practice of not requiring the OL stage equipment to meet the latest safety requirements because the project is now considered to be out of date, "cast in concrete," and impossible to change at this late stage. New criteria are waived, exceptions are made, and shortcuts are taken in the desire to get the plant operating, as it represents a large investment and a needed energy source.

Various proposals have been made by both nuclear promoters and nuclear critics calling for a one-step licensing process. This process would require that the detail design be completed and verified before the construction permit is issued. Since the design could be reviewed in depth, subsequent operating license review would consider only the utility's qualifications and procedures for operating the plant, and would not address issues of design adequacy unless new information concerning previously unknown problems is received. It is strongly recommended that serious consideration be given to such improved design reviews prior to initiation of plant construction. The current practice of allowing "obsolete" or "sub-standard" designs to go into operation has resulted in degradation of plant safety and is a process should not be permitted to continue.

4.6.2: Generic Reviews and Regulatory Guides

In the review conducted by the NRC in the licensing process, substantial reliance is placed on references of similarity to plants previously approved, to system designs that have been generically reviewed, and to configurations that have been described as adequate for licensing purposes in non-mandatory Regulatory Guides. This relatively hodge-podge system of design documentation makes it extremely difficult to determine what actually is being proposed for construction or operation and makes it near impossible

to critically review the actual configuration that will result out of the design process. Although the licensing process is theoretically open to the public, a complete review of all documents actually contained in any one license docket will not include a substantial portion of the descriptive information actually applicable to the particular plant. This is a problem not only for the public participant, but for the individuals with actual design and operating responsibility throughout the life of the plant. Uncertainty as to the "as-built" configuration, interchangeability of renewal parts, and other details required for failure or accident evaluation result in less than acceptable safety results. It is recommended that improvements be made in the licensing process documentation requirements such that not only is the proposed plant design completely described, but it is also updated periodically so that the descriptive documentation is complete and accurate.

4.6.3: Legalistic vs Technical Aspects of Review

The licensing review of a proposed nuclear plant is often thought of as the technical review to ensure design and system adequacy to operate safely and efficiently. Unfortunately, the licensing process focuses primarily on the legalistic aspects of the review and it in general degrades to a contest of determining the minimum required to meet the letter of the law. Public participation in this review process is invited and

described as essential, but in actuality intervenors in the licensing process are expected to prove that the system is unsafe, rather than having the applicant show that the system is safe. This is perhaps a carry-over from the U.S. system of justice wherein the defendant is assumed innocent until proven guilty. In the case of licensing nuclear power plants, this attitude places a near-impossible burden on those individuals attempting to raise questions of design adequacy. Technical contentions are argued primarily on their legalistic merits, and contention content is continually narrowed to the point of determining only if the word of the regulation is met. Contentions that challenge the adequacy of the regulations are not considered in the normal licensing process, as these are interpreted to be beyond the authority of the licensing board.

It is difficult to determine a way of overcoming this deficiency in the licensing process. Stability, and a design freeze is obviously essential if the project is to be actually built. On the other hand, design evolution and changing safety standards must be a part of the process if safer LWR's are to be produced. Some way of improving the effectiveness of the licensing process should be possible if improved safety is given a high priority.

4.6.4: Documentation of Deviations

It has been standard practice to date for the NRC to determine that it is not necessary for a number of items in a specific plant to meet the current NRC practices and safety requirements. It has been recommended in the past that the NRC's Safety Evaluation Report (SER) should include a summary in which the pros and cons that have entered into the decision to exempt the requirement are documented. The failure to provide such a "documentation of all deviations" to current NRC practices in the plant-specific SER is a recognized regulatory weakness. For example, the NRC, in 1976, instituted a plan to document departures from the Standard Review Plan.^{15/} While the NRC later drew back from this position and failed to implement these procedures, these directives by NRC management underscored the need to have on record in the SER the factors entering the Staff's analysis of facility safety. Absent such a listing, it is nearly impossible for an independent party to conduct an informed analysis and inquiry into any safety impacts these deviations may have. A complete listing and evaluation of potentially significant safety items which do not comply with present regulatory requirements is imperative for a definitive assessment of plant safety.

4.6.5: Cumulative Impact of Deficiencies and Deviations

A weakness in the current implementation of the NRC regulations is that the NRC does not quantitatively evaluate

the cumulative impact of all the safety issues under consideration. The implementation of new NRC branch positions, new regulatory guides, and new issues that arise in licensing case reviews are each evaluated individually. Little or no attempt is made to look across the board at the composite impact of all the accumulated deficiencies. In determining the need for backfitting operating plants and those with construction permits, the NRC should be required to quantitatively assess the cumulative impact of the sum of safety hazards that are unresolved, under consideration, and grandfathered.

4.6.6: Backfitting

Evaluation of non-compliance with current regulatory practices, and of the safety impact of unresolved generic problems may indicate the need for modification or backfitting which would provide "substantial additional protection which is required for the public health and safety." (10 CFR 50.109) Obviously, backfitting decisions made after construction has been initiated may be more expensive to implement and may be effectively precluded by cost and difficulty of revising existing plant structures, systems, and components. Until supplemental safety analyses are carried out by the NRC in the SER, an adequate assessment of required modification and backfitting is precluded. Additionally, NUREG-0410 defines outstanding safety issues classified as Category A as matters which, if resolved, would, in part ".....

provide a significant increase in the assurance of the health and safety of the public." ^{16/} As a minimum, all relevant generic Category A items should be addressed by the NRC and the board when backfitting requirements are assessed. Additional items to be considered are those items identified in NUREG-0510 as unresolved safety issues, and the many recommendations that will be issuing from the Three Mile Island reviews.

In addition, the NRC procedures and policies for regulating operating plants should be reassessed and strengthened. For example, the backfitting regulation might be amended to correct the present reversal of roles between the plant owner and the NRC. In the revised roles, the plant owner should be required, as a matter of course, to propose plant and procedural modifications, if any, in response to all new NRC regulatory requirements, while the NRC should only have the responsibility for assessing the adequacy of the plant owner's response. If the plant owner determines that no changes to the plant are required by the new regulations, this fact should also be reported and the appropriate rationale presented. A requirement for the plant owner to deal with unresolved safety and generic issues in this manner would serve to prevent the buildup of numerous generic lists which go on and on from year to year without resolution.

4.6.7: Licensing Conclusions

- In conclusion, it appears that a fairly dramatic overhaul is required to improve the effectiveness of the licensing

process. Having been intimately involved with this process for a number of years, we cannot help but being drawn to the conclusions that the licensing process is a legalistic game and as currently conducted provides little assurance of technical safety review. It is strongly recommended that a multi-disciplinary group be assembled to try and develop a workable process which has achievement of safety, rather than licensing of plants, as its primary objective.

4.7: OTHER

In categorizing the various safety assurance effectiveness issues, two were identified with broad applicability or which did not seem to fit totally under any of the five preceding categories. These broad issues are addressed as follows:

4.7.1: Periodic Plant Reviews

After completion of the operating plant license review, no comprehensive review is mandated by the regulations as the plant operation continues and the unknown aging effect occurs. The NRC has implemented on an ad hoc basis a program called the "Systematic Evaluation Program" (SEP) to evaluate the first eleven plants licensed. This program has yet to really get under way and its completion has been severely delayed by the reassignment of key personnel to the Three Mile Island reviews. Three Mile Island undoubtedly should receive top priority, but the development of procedure and requirements for periodic plant

reviews such as those envisioned by SEP should be given a top priority by the NRC. This may require the development of new methods of assessing the effects of aging and radiation level buildup, and certainly requires a formalization through the regulations and development of standard reporting and documentation format. It is most likely that the very newest and the very oldest plants represent the most significant contribution to the risk of the total nuclear plant program. This deficiency should be expeditiously corrected.

4.7.2: Constitution of Review Boards

This issue may well fit under the general category of licensing process but because the concern and recommendation is of relatively far-reaching significance, it is handled here separately. In the ongoing review of licensing and safety issues, three separate review board constituencies play a major role in the assessment and achievement of safety. These three bodies are those persons who constitute the Atomic Safety and Licensing Boards, the members of the Advisory Committee on Reactor Safeguards, and the Plant and General Office Review Boards provided at each operating plant to review the safety aspects of plant operation and modifications. In theory, these bodies or boards provide an independent assessment of the designs, or of the plant-unique attributes to meet the regulatory requirements on which their operation is based. One common failing exists for

all of these bodies. They are not really independent from the organizations which they are assembled to evaluate.

For example, the members comprising the Atomic Safety and Licensing Boards are for the most part permanent employees of the NRC. This does not mean that they cannot make unbiased decisions, but it does present one possible impediment to the achievement of truly objective decisions. The ACRS is comprised of individuals theoretically independent of the NRC and, in fact, a large percentage of the ACRS members are selected from the academic community. However, their ACRS duties are not full-time positions and most members of the ACRS are closely tied to the nuclear establishment through other interests. In addition, the ACRS members are heavily dependent upon a full-time staff which is a part of the NRC organization. Similarly, the plant-unique General Office and Plant Review Boards are comprised primarily of utility employees from the various technical divisions, all having the possible impediment of conflict of interest and job bias potentially affecting their decisions.

The various review and advisory board functions can be considered philosophically to be a quality assurance type function in that open and objective evaluation of the facts is essential. It is interesting to consider the organizational requirements specified by Criterion I of 10 CFR 50 Appendix B for quality assurance functions. This criterion specifies

that:

"The persons and organizations performing quality assurance functions shall have sufficient authority and organizational freedom to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions. Such persons and organizations performing quality assurance functions shall report to a management level such that this required authority and organizational freedom, including sufficient independence from cost and schedule when opposed to safety considerations, are provided." 17/

As a practical matter, it is generally impossible to totally implement this freedom from cost and schedule considerations in the day to day quality assurance organizations. However, it would appear that in the constitution of critical review boards and bodies, extra precaution should be taken to adhere to this necessary attribute. It is recommended, therefore, that members of the Atomic Safety and Licensing Boards should not be employees of the NRC. Advisory Committee on Reactor Safeguards appointees should be carefully selected and screened so as to preclude bias, and permanent ACRS staff should be disassociated from the NRC. Additionally, ACRS recommendations should be given more than an "advisory status" and actions or decisions made by the NRC that conflict with the ACRS position should require extraordinary consideration. Utility General Office Review Boards should include one or more members from outside the utility, preferably

from state or local government agencies and General Office Review Board meetings should be subjected to some of the "sunshine" currently prescribed for NRC meetings.

4.8: CONCLUSIONS

In conclusion, the importance of satisfactorily addressing the deficiencies discussed in this Section 4, Safety Assurance Effectiveness, cannot be overemphasized. It may do little or even absolutely no good to spend hundreds of millions of dollars and countless resources to develop and test the best safety system in the world if it can be bypassed by an improperly qualified operator, or defeated by a forgotten procedure. Light water reactor safety improvements in probability reduction and in consequence mitigation are meaningless if those improvements are not actually achieved in the hardware and systems that comprise the nuclear plant. Because the problems identified in this section are for the most part human failings, or at least produced as a result of human omissions or commissions, it is more difficult to identify research and development programs that may serve to resolve these problems. The Three Mile Island-initiated reviews provide a starting point for the development of answers to many of these questions. The impetus of TMI should be utilized to expand the human interface task forces beyond the confines of human engineering, of control rooms, and operator qualification to cover all of the safety assurance issues addressed herein.

This will require the development of an extensive and complete safety assurance program which would go beyond the paper requirements that are often mistaken for quality assurance. A summary of the recommended programs developed from these safety effectiveness issues is contained in Section 6.1.

SECTION 4

REFERENCES

- 1/ Testimony of Richard B. Hubbard and Gregory C. Minor Before the Federal Trade Commission Regarding Standards and Certification, Proposed Rule 16 CFR Part 457, May 18, 1979.
- 2/ EMD-77-30, Allegations of Poor Construction Practices on the North Anna Nuclear Powerplants, U.S. General Accounting Office, Washington, D.C., June 2, 1977
- 3/ "Browns Ferry Nuclear Plant Fire," hearings before the Joint Committee on Atomic Energy, September 16, 1975.
- 4/ NRC Press Release No. 76-122, entitled "Independent Assessment of NRC Quality Assurance Activities Planned," May 25, 1976.
- 5/ NUREG-0321, A Study of the Nuclear Regulatory Commission Quality Assurance Program, U.S. Nuclear Regulatory Commission, Washington, D.C., August, 1977.
- 6/ EMD-78-80, The Nuclear Regulatory Commission Needs to Agressively Monitor and Independently Evaluate Nuclear Power Plant Construction, U.S. General Accounting Office, Washington, D.C., September 7, 1978.
- 7/ Examples include:
 - A. NUREG-0397, Revised Inspection Program for Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Washington, D.C., March, 1978.
 - B. NUREG-0425, NRC Inspection Alternatives, U.S. Nuclear Regulatory Commission, Washington, D.C., February, 1978.
- 8/ Minutes of Meeting Two of the Risk Assessment Review Group, U.S. Nuclear Regulatory Commission, Washington, D.C., p. 4.
- 9/ NUREG-0578, TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations, U.S. Nuclear Regulatory Commission, Washington, D.C., pp. A-57 and A-58.
- 10/ NUREG-0600, Investigation Into the March 28, 1979 Three Mile Island Accident by Office of Inspection and Enforcement, Investigation Report No. 50-320/79-10, U.S. Nuclear Regulatory Commission, Washington, D.C.

- 11/ EPRI-NP-1118-SY, Project 501-3, Human Factors Methods for Nuclear Control Room Design, Electric Power Research Institute, June, 1979.
- 12/ Ibid 10, p. I-2-50-53.
- 13/ Draft Regulatory Guide and Value/Impact Statement, Radiation Protection Training for Light-Water-Cooled Nuclear Power Plant Personnel, August 1979, Division 8, Task OH 717-4.
- 14/ Ibid 10, p. I-3-6.
- 15/ NRC Office Letter No. 9, Attachment F, September 20, 1976, Memoranda, "Documentation of Deviations From the Standard Review Plan."
- 16/ NUREG-0410, NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants, Appendix B, January 1, 1978.
- 17/ Title 10, Code of Federal Regulations, Section 10, Appendix B, Criterion I.

SECTION 5

PROGRAM PRIORITY EVALUATION

5.1: GENERAL DISCUSSION

As the preceding sections well illustrate, there is no lack of possible safety improvement programs for which to utilize future research and development efforts. Perhaps the converse is more likely; programs identified have in the past been so diverse and numerous that research and development efforts have suffered from the lack of direction and focus. In addition, the "it can't really happen syndrome" has clouded the decision process to the extent that only research and development programs of general scientific interest have been emphasized. Other safety improvement programs have been resisted by the industry because of the inherent backfitting implications, so improvements are effectively stalled by lengthy consideration of "how safe is safe enough" arguments (i.e., ATWS). Allocation of resources to resolve some issues has been rejected because the improvement expected was thought to be insignificant due to the low probability of the postulated event. Three Mile Island has changed much of this. The safety research program mandated by Congress to the NRC and DOE, coupled with resource shortages (manpower) due to the TMI reviews, give rise to the need for prioritization and selection of the most important programs out of the many available.

The question of which of the programs should be allocated the finite and limited resources, or, the selection of alternatives, is a common problem to both the government and private industry. Priority analysis and selection occurs in any budgeting process and is subject to the political pressures of both budget cutting and empire building. In the past, the safety research program developed first by the AEC, and subsequently the NRC, has often been subject to an emphasis of programs needed to promote nuclear power. Even as recently as 1978, the NRC's Annual Report for that year states the research objectives of the NRC are "to define with greater precision the safety margins provided in nuclear facilities,"^{1/} a subtle implication that safety should be reduced, not improved. Further, one of the five programs approved through the 1978 safety program was "improved seismic design." This program is not needed to make plants safer, it is primarily to make them licensable at less than desirable sites.

As a part of the Nuclear Regulatory Commission authorization for appropriation for fiscal 1978, Congress amended the Energy Reorganization Act of 1974, requiring the NRC to prepare a long-range plan for the development of new or improved safety systems for nuclear power plants.^{2/} The affect of the Congressional mandate was to move the NRC's research program planning into a more formal process, with safety improvement a specific objective.

This requirement has resulted in additional research and development assistance being requested by the NRC through the Department of Energy and its associated contractors. The increased attention on research program planning has resulted in the issuance over recent years of several documents of relevance in determining priority of programs for allocation of resources. These include:

1. The 1977 ACRS review of the NRC's proposed 1978 safety research budget (NUREG-0392). 3/
2. The 1978 ACRS review of the NRC's proposed 1979 safety research budget (NUREG-0496). 4/
3. The NRC's Plan for Research to Improve the Safety of Light-Water Nuclear Power Plants, April 12, 1978 (NUREG-0438). 5/
4. The update of the NRC's plan for research to improve safety, included in the NRC's 1978 Annual Report, Chapter 11. 6/

These reviews and assessments have all utilized informal application of cost/benefit methodology to attempt to determine the largest payback for the resources to be expended. However, the cost/benefit analysis technique is only as accurate as the ability to predict all costs and benefits, and to develop suitable methods for handling the uncertainty and non-quantifiable portions of the equation. In the highly speculative area of nuclear safety improvement, only the research costs can be reasonably estimated and the risk avoidance, or benefit, is near impossible to calculate. Accordingly, safety improvement priorities have

to date, been largely subjectively set by those in positions of responsibility. Attempts have been made in the past to more systematically set priorities for extensive lists of diverse tasks. After struggling for years with the generic issues compiled through the ACRS lists, the Technical Safety Activities Report, and other sources of unresolved safety items (see Section 2), the NRC assembled this "amorphous group" of concerns into the 133 generic issues published in NUREG-0410.^{7/} A minimal attempt was made to establish priorities which resulted in the assignment of the issues into Categories A through D.^{8/} An additional step was taken by the NRC in 1978 in the attempt to further ascertain priorities for the 133 generic issues. This step was to utilize risk assessment methodology in the identification of top priority generic issues. The use of this methodology is briefly described in the NRC's 1979 report on generic issues,^{9/} and resulted in the identification of a new priority list of "17 Unresolved Safety Issues."^{10/}

The lack of real progress towards resolution of these items however, was acknowledged by a comment of one ACRS member during the Staff's October 6, 1978 presentation in which he stated that the NRC seemed to be doing a fine job of managing and recompiling lists but that very little progress seemed to be made towards getting the necessary work done to reduce the lists. Even with little apparent progress toward completion,

these steps towards a more scientific ranking of safety priorities seemed encouraging. The program was then almost totally upset by the Three Mile Island accident. The nuclear industries' new focus on TMI issues will almost certainly result in a drastic realignment of priorities. This illustrates clearly a failing of the nuclear program to date. Priorities have not been rationally planned, they have been reactions to outside pressures. This fact appears in NRC Commissioner Bradford's recent speech on regulatory effectiveness in which he concedes that "the NRC's role in the past has been 'heavily reactive.'" 11/

It is most likely all will concede that an accepted and effective mechanism for establishing safety research priorities does not yet exist. The following section will present ideas and recommendations on overriding considerations that should be used in establishing priorities, and how the selection process can be improved.

5.2: PRIORITY EVALUATION

As a starting point, the selection methodology used in the NRC's first plan for safety improvement 12/ has some very useful ideas. The method used was to start out with a large list of proposed programs gathered from the various sources identified in the report. These several hundred proposed programs were categorized into 16 different research

topics which were subsequently reduced to 5 projects recommended for action. This selection was accomplished through the means of a qualitative selection process based on judgmental assessment of 4 factors: 13/

1. The breadth of support for the project (agreement by knowledgeable persons that effort is needed).
2. The assessed risk reduction potential.
3. Generic applicability.
4. Anticipated cost of implementation.

The assessment performed was a preliminary rough cut wherein Factor 1 was a "go/no-go" decision. Factors 2, 3, and 4 were then subjectively ranked as high, medium, or low. This method clearly is a qualitative type cost/benefit analysis, and if adequately expanded and refined, would probably be the most logical way to arrive at priorities. A pictorial concept of such a refinement is the multi-stepped decision process flow-charted in Figure 5-1.

The advantages of this process are obvious; important issues are immediately elevated to action status. Lesser issues flow through the cost/benefit process and are evaluated for incremental safety impact, a step not now being done.

In essence, to perform an accurate cost/benefit analysis, what is needed is an accurate assessment of Factors 2, 3, and 4 of the four safety plan factors listed above.

FIGURE 5-1

SIMPLIFIED SAFETY ISSUE EVALUATION

This means that it must be possible to:

- Quantify the risk reduction potential.
- Determine the generic applicability.
- Calculate the cost of implementation.

All of this must be done with great care, taking into account all of the direct and indirect effects. Factors that are important to the prioritization process and which should help determine the programs to be selected include, but are not limited to:

TABLE 5-1

FACTORS IMPORTANT TO SAFETY ISSUE EVALUATION

1. Primary improvement must be greater than the secondary risk added.
2. Compatibility with the requirements of the General Design Criteria.
3. Quantification of cost/benefit possible without extraordinary uncertainty.
4. Applicability to the largest number of plants.
5. Applicability to plants which could potentially affect the largest population segment.
6. Applicability to plants where accidents could impact on valuable or irreplaceable resources.
7. Applicability to plants containing systems or components that have been found by experience or analysis to have critical design faults.
8. Complex issue programs should develop diverse solutions (improve both probability and consequence aspects).

9. Should be focused so as to address a specific problem such as:
 - a. Class 9 accidents.
 - b. On-site consequences.
 - c. Occupational dose reduction.
 - d. Etc.
10. Should bring risks below the level of an agreed upon threshold criteria (develop success criteria).
11. Should be applicable to currently operating versus future plants (there are likely to be more of the former).
12. Should be of low cost implementation (if possible).
13. Should result in early payback (benefit).
14. Should be one that will work (pragmatic).

As the above factors are further evaluated and studied, it becomes apparent that some are directly conducive to analysis. For example, demographic data already is available to rank plants in order of potential accident health effects. A logical step would be to evaluate the worst case demographic plants and attempt to identify commonalities between them for possible high priority safety improvement programs.

The scope of this study has not included the detailed consideration of such factors. It is understood that Sandia is conducting such decision process studies. These should be continued at a high level of priority in order to assist in evaluating resource allocation.

5.3: SAFETY PROGRAM SELECTION RECOMMENDATIONS

In our evaluation of safety program selection criteria, we have arrived at a number of conclusions which are recommended for general guidance in the selection process. These conclusions are based both on past experience and on an intense and lengthy period of active participation in public safety reviews. It is therefore recommended that the following conclusions be given substantial weight by the authorities in determining the future course of the safety research program:

- Research program priorities must recognize that accidents can happen. Thus, consequence mitigation projects, such as increased understanding of radiation migration through liquid and soil pathways, should be candidates for receiving greater emphasis. In addition, consequence mitigation expenditures also seem to be an appropriate area of federal funded research, as opposed to nuclear industry funding, since consequence mitigation measures may cut across numerous local and state jurisdictions. Finally, even if the plant owner or the reactor supplier wished to fund consequence mitigation research, it is clear that much of the resultant effort would of necessity occur off the plant site, and hence beyond the control and jurisdiction of the plant owner.

- Funding sources for research addressing accident prevention measures should be re-evaluated. To a greater extent, design verification testing of the application of LWR structures, systems, and components should be funded by the nuclear suppliers and the reactor owners. The NRC, rather than conducting research programs that parallel industry efforts, should consider increasing their "hands-on" participation in the industry's design verification programs. As an example of increased participation, NRC personnel should both approve industry test plans and witness the resulting tests. The present NRC program of reviewing test results should continue, but with the added knowledge gained as a full participant in the test plan development and implementation. The roles of the NRC and the nuclear industry on computer model verification should also be re-assessed. For instance, the nuclear industry should fund programs such as LOFT that are used to verify industry computer codes. NRC may develop simplified computer models, but the nuclear industry should retain the primary responsibility for developing design verification data to document the adequacy of the computer models in the application and design of LWR structures, systems, and components.

- Programs that result in improved safety of operating plants should be given a high priority in the reassessment of the LWR safety research program.

In addition, the NRC procedures and policies for regulating operating plants should be reassessed and strengthened. For example, the backfitting regulation (Title 10, Part 50.109) should be amended to correct the present reversal of roles between the plant owner and the NRC. In the revised roles, the plant owner should be required, as a matter of course, to propose plant and procedural modifications, if any, in response to all new NRC regulatory requirements, while the NRC should only have the responsibility for assessing the adequacy of the plant owner's response*. If the applicant determines that no changes due to the new regulations are necessary, this fact should also be reported and justified. For a second example, the NRC presently has no record of the degree of compliance of operating plants with current regulatory practices-- e.g., regulations, regulatory guides, branch technical

* Presently, in accordance with the 10 CFR 50.109, the NRC is given the responsibility of making the initial determination of whether backfitting to provide substantial, additional protection is required for the public health and safety or the common defense and security.

positions, and the Standard Review Plan. The NRC should initiate a program immediately to prepare and issue supplementary Safety Evaluation Reports (SER) for all operating plants which document all deviations from current regulatory practices. Documentation of each plant's present adherence to regulations by periodic updating of the FSAR would thus form the foundation of any program to improve the safety of operating plants. For a third example, increased NRC attention should be given to developing and implementing an effective and disciplined quality assurance program for the maintenance and modification of operating plants. Collection and analysis of field failure data and licensed event reports should be an integral part of each operating plant's quality assurance program. Further, the quality assurance program should assure that the safety of the plant is not degraded as a result of maintenance activities or plant modifications. Another aspect of operating plant safety is that one can postulate a serious radiation release accident may be more likely during reactor shutdown than during plant operation. The WASH-1400 risk assessment did not formally analyze the probability and consequences of potential accident sequences which might occur

during plant shutdown with the exception of the analysis of an accident in the spent fuel pool. The omission of an analysis of shutdown accidents is a potential weakness in WASH-1400. During plant shutdown, some levels of the defense in depth may be partially or full inoperable, a number of maintenance personnel may be located throughout the facility, and the alarm status indicators in the control room may be both rich with alarms and also in a condition not considered during operator training. Therefore, the quality assurance program discipline is particularly important during plant shutdown

- There are roles that federal authorities can uniquely fulfill in the area of accident prevention. For example, research to define most externalities which might impact on the plant site, such as floods, earthquakes, tornadoes, winds, hurricanes, etc., are an area where federal government involvement appears justified. As an added benefit, any information that the federal government develops defining externalities, can also be utilized for guidance in siting other industrial and residential facilities. The

WASH-1400 authors have acknowledged their lack of a complete understanding of the physical processes occurring during a major reactor accident. Research to develop data on these physical processes (such as fuel meltdown and migration) appears to be an appropriate area for federal funding.

- Cost effectiveness may be a key consideration in the determination of LWR safety program priorities and focus. Experience to date would indicate that LWR safety programs that result in improved plant reliability, as well as quality, will be cost effective. For example, plant capacity factors have averaged over time in the low "60's." Improvements in capacity factors to the mid-"80's" could result in both reduced operating costs and a reduction by one-third in the number of nuclear plants constructed in order to obtain a given number of kilowatt hours of nuclear capacity, thus statistically reducing the risk to the public. The TMI-2 accident vividly demonstrates the extreme financial penalty caused by long plant outages and/or early plant shutdowns. Cost/benefit analyses should give proper consideration to both the benefits of improved plant capacity as well as to the risks of extended outages and/or early plant shutdown.

SECTION 5

REFERENCES

- 1/ U.S. Nuclear Regulatory Commission, Annual Report 1978, p. 180.
- 2/ NUREG-0438, Plan for Research to Improve the Safety of Light-Water Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Washington, D.C., April 12, 1978, p. 1.
- 3/ NUREG-0392, Review and Evaluation of the Nuclear Regulatory Commission Safety Research Program, U.S. Nuclear Regulatory Commission, Washington, D.C., December, 1977.
- 4/ NUREG-0496, Advisory Committee on Reactor Safeguards, 1978 Review and Evaluation of the NRC Safety Research Program.
- 5/ Ibid 2.
- 6/ Ibid 1, Chapter 11.
- 7/ NUREG-0410, NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Washington, DC, January 1, 1978.
- 8/ Ibid 7, Appendix B.
- 9/ NUREG-0510, Identification of Unresolved Safety Issues Relating to Nuclear Power Plants, U.S. Nuclear Regulatory Commission, Washington, D.C., January, 1979.
- 10/ Ibid 9, p. 6.
- 11/ Remarks by NRC Commissioner Bradford before the NARUC Annual Regulatory Studies Program, East Lansing, Michigan, August 2, 1979.
- 12/ Ibid 2.
- 13/ Ibid 2, pp. 8 & 9.

SECTION 6

CONCLUSIONS AND RECOMMENDATIONS

6.1: SUMMARY OF RECOMMENDED PROGRAMS

Following is a summary of recommended programs developed from the information contained in Section 2 (Probability), Section 3 (Consequence), and Section 4 (Safety Assurance). In some of these summaries, indication of urgency or high priority will be found. The purpose of this is merely to identify some of the higher priorities recommended within a given section and should not be taken as the final priority evaluation for this total study. For the overall discussion of priorities for the safety research programs to be conducted by or for the federal government, refer to Section 6.2.

6.1.1: SAFETY RESEARCH PROGRAMS TO REDUCE ACCIDENT PROBABILITY

The NRC, ACRS, and other collegial groups have generated lists of reactor safety issues, most of which are still classed as "unresolved." The TMI post-accident analyses have created several more lists of safety issues requiring correction. Few of these issues are concerned with the mitigation of accident consequences; most concern accident probability. Almost none of the issues address "safety assurance effectiveness." Most issues included on the formal lists are not discussed extensively here as substantial work is already underway; one major finding, however,

is that such lists have existed for years and what is needed now are decisions, solutions and commitment to their implementation. Creating a list does not solve the problem. The following probability-related programs are additionally recommended for particular emphasis:

- HUMAN FACTORS RESEARCH (SECTION 2.5.1)

Today's reactor control rooms are generally antiquated and poorly designed from a human factors point of view. There is an urgent need for a human factors research facility to evaluate control room changes and computer-based control/display additions, and to provide hard data on their impact on operating effectiveness and safety.

- AUTOMATION vs MANUAL PROCEDURES (SECTION 2.5.2)

Post-TMI recommendations are directed to increase automation of safety features and also to rely more heavily on operating procedures and operator's innovation as a part of the defense-in-depth. Before any major shift is made, it is important to research the impact on safety of the two alternatives. This study should also consider a 10-minute versus 30-minute (or longer) rule for operator inaction during an accident.

- ADVANCED SIMULATORS (SECTION 2.5.4)

Operator training should be expanded by using more advanced simulator models covering more transient and upset conditions. Research is needed to develop these more sophisticated models. The study should also look at the optimum retraining period.

- ATWS ALTERNATIVES (SECTION 2.5.3)

The ATWS accident is a high consequence accident which warrants prevention measures. It has been deadlocked for ten years and would benefit from independent research and recommendation of alternatives.

- ADDING A REACTOR "FLIGHT RECORDER" (SECTION 2.5.5)

TMI had the benefit of a multichannel recorder to aid in the evaluation of the accident causes and sequences. Research should be conducted to decide on an optimum set of parameters to measure, and appropriate design requirements. The reactor "flight recorder" should then be made a requirement on each operating plant.

- REACTOR SITING ALTERNATIVES (SECTION 2.5.6)

The jurisdictional restraints of utilities has resulted in some marginal to poor sites for nuclear plants. Rather than work on the ways to permit operation at these poor sites, studies should be made to find good sites and figure out effective distribution of power to areas which have poorer sites.

- POST ACCIDENT MONITORING (SECTION 2.5.7)

The present instruments for measuring accident and post-accident conditions are generally inadequate. Research may be needed to insure that instruments are available to track and evaluate post-accident conditions.

- SAFETY/CONTROL INDEPENDENCE AND CLASSIFICATION (SECTION 2.5.7)
There has been a problem of lack of independence of safety/control functions, mainly in PWRs. Accidents have been experienced where control systems were relied upon for safety of the plant. Studies are needed to determine the need for control systems in safety situations and to evaluate the reduction in safety from lack of separation.
- PROVIDE DIRECT LEVEL MEASUREMENT ON PWR VESSEL (SECTION 2.5.7)
The operators in PWRs have no direct reading of vessel level. Research is needed to determine alternate methods of measuring the water level in the core.
- COMMON-CAUSE FAILURES AND SINGLE-FAILURE CRITERIA (SECTION 2.5.8)
TMI has cast doubt on the adequacy of the single-failure criteria and shown the need for greater attention to common-cause failures. Both of these issues need to be researched in terms of possible safety improvement.

6.1.2: CONSEQUENCE MITIGATION PROGRAMS

Any calculation of the absolute probability of core melt accidents and the resulting accident prevention measures, will contain substantial uncertainties. Therefore, LWR safety research must also provide answers on how to mitigate the consequences of routine and accidental radioactive releases. Consequence mitigation programs recommended in this Study include the following:

- MODEL DEVELOPMENT-SITE SPECIFIC CLASS 9 ACCIDENT CONSEQUENCE ANALYSIS CODE (SECTION 3.2)

The NRC's NEPA environmental reviews have not included calculations of the consequences of Class 9 accidents. As a result of the Class 9 accident at TMI-2, and the Risk Assessment Review Group's conclusion that estimates of the absolute probabilities of accidents in WASH-1400 are not reliable, the NRC no longer has a theoretical or practical basis to justify excluding the environmental assessment of Class 9 accidents. The consequence (CRAC) model developed for WASH-1400 was developed to estimate aggregate societal risks and thus contains many simplifying assumptions that introduce substantial uncertainties into any site specific calculation. Safety research programs are recommended to complete the necessary analyses and experimental verification to reduce the uncertainties cited in the CRAC Code.

- SOIL AND LIQUID PATHWAY ANALYSES/INTERDICTION TECHNIQUES (SECTION 3.3)

Models to evaluate the soil and liquid pathway following a core melt-through accident, have not been developed with the same comprehensiveness as for atmospheric releases. The financial costs and the potential societal dislocation due to water contamination from strontium-90 and other isotopes appear to be potentially very large.

The analytical models deserve continued development. In addition, installing the necessary interdiction systems as a standard part of the safety equipment of each nuclear power plant should also be evaluated.

- LOW-LEVEL WASTE MANAGEMENT (SECTION 3.4)

The NRC faces immediate challenges relative to setting criteria for the disposal of low-level wastes, for the management of uranium mill tailings, and for the handling of spent fuel. The current NRC research program does not appear to reflect the relative urgency with which each of these problems needs to be addressed. Research programs in the following areas are identified: methods for disposal of low-level wastes, ground water hydrology, methods for assaying wastes, management of gaseous wastes, and methods for effecting the removal of radionuclides deposited internally.

- LAND CONTAMINATION CRITERIA DEVELOPMENT (SECTION 3.5)

The potential consequences of a major nuclear accident are extremely far-reaching in terms of both time and geography. Research should be conducted to enable federal authorities to develop criteria for the appropriate threshold for interdiction or decontamination. Also, research to enable federal authorities to establish nationally and internationally accepted land

contamination criteria, plus international standards on general population allowable dosage should be included in the LWR safety research program.

- DECONTAMINATION METHODS AND POTENTIAL COSTS (SECTION 3.5.1)

The technical and economic feasibility of decontaminating land, buildings, and pavements exposed to releases of radioactivity in a reactor accident has not been demonstrated. Such studies should be completed on a high priority.

- EMERGENCY RESPONSE PLANNING (SECTION 3.6)

Implementation of an effective emergency response plan could significantly reduce the prompt health effects resulting from an accidental radioactive release. The emergency response to the TMI-2 accident once again teaches that planning efforts are not adequate. Aspects of emergency planning which should be addressed in the safety research program include accident size, automated alarming and initiating systems, adequacy of health treatment facilities, local population growth and siting criteria, and public training measures.

- RADIATION EXPOSURE OF PLANT EMPLOYEES (SECTION 3.7)

The operation, maintenance, and retrofit of nuclear plants is requiring ever increasing exposure of plant personnel due to the increasing levels of radiation within the

plants, and the increase in plant numbers and size. A comprehensive program to address all aspects of this problem, and to establish requirements to insure that the future health impact is minimized is required. The research program should include consideration of plant configuration, material selection, shielding, automated tools, plant decontamination and decommissioning, and the acceptability of current exposure limits. The impact on plant operability, if individual exposure limits are reduced, should also be evaluated.

6.1.3: SAFETY ASSURANCE PROGRAMS

In this vital area of programs needed to assure the effective implementation of safety in light water reactors, six general areas were considered. From these, we have identified specific needs which should be given emphasis in the federal safety improvement plan. Some will require research and development; others are more in the nature of policy considerations. All are considered of high priority. Where "normal" research programs can address these needs, those programs should be initiated. Where policy, or quality of implementation issues are identified, it is recommended that multi-disciplinary teams be assembled to develop comprehensive programs to be carried out as a part of the safety program. The following programs are recommended:

- DESIGN EFFECTIVENESS (SECTION 4.2)

The nuclear plant design process has grown out of a complex mix of practices and requirements that are not smoothly integrated. Specific program needs are to:

- Further assure design responsibility by keeping the designer in the loop throughout plant life.
- Perform a thorough review and update of the General Design Criteria.
- Improve the quality and timeliness of the Code and Standards production process, and strengthen compliance requirements.
- Improve the design verification process through mandated FMEA and greater reliance on timely and comprehensive tests. Tests should satisfy the needs for environmental qualification and aging effects.
- Expand the field feedback system and require a documented system of failure disposition notices.

- QUALITY ASSURANCE IMPLEMENTATION (SECTION 4.3)

The implementation of the general requirements of the quality assurance criteria has been ineffective. Recommendations are to:

- Make changes previously identified in past reviews and audits.
- Require more careful and timely updating of "as-built"

drawings and FSAR, and

- Determine how to factor quality assurance deficiencies into risk assessment.

- REGULATORY ENFORCEMENT (SECTION 4.4)

The methods and authority of the NRC have not resulted in adequate enforcement of regulatory requirements. A program should be implemented to bring about the following improvements or changes:

- The quality and frequency of licensee inspections and audits should be upgraded. This should include more frequent and extensive supplier inspections.

- Resident inspectors should be assigned to all plants and at the facilities of major suppliers.

- Penalties for non-compliance should reflect the seriousness of the potential consequences and should accrue to utility management or shareholders, not the ratepayers.

- Dissenting views should be handled with more objectivity, and periodic audits of the NRC should be required.

- PERSON-MACHINE INTERFACE (SECTION 4.5)

A broad range of failures have occurred ranging from the inability of the operator to comprehend the critical information displayed by the machine, to more straightforward errors in procedure, management, and performance of assigned responsibility. This human interface should

receive intensive study and improvement through a program that addresses:

- Improvement of human factors design and standards development, including investigation of backfitting requirements.
- Development of standards for judging adequacy of simulator training, including length and frequency of training, and improvement of abnormal operation response training.
- Evaluation of tradeoffs between automation and manual response and the setting of appropriate standards.
- Consideration should be given to the training and licensing of currently non-licensed utility operators, maintenance personnel, management level persons, and the training or control of contract personnel.
- Additionally, studies should consider the susceptibility of the plant and equipment to sabotage or intentional misoperation.

● LICENSING PROCESS (SECTION 4.6)

A number of problems have been identified in the licensing process that serve to lessen the effectiveness of the safety review. In addition, the licensing process does not adequately address the ongoing safety

reviews needed for the operating plants. Accordingly, it is recommended that effort be applied to develop:

- An improved method for handling the review of design associated with the construction permit application.

This should consider the problem of lack of design detail at that stage and the reverse problem of completed construction at the operating license stage.

- Improvements should be made to the documentation procedures, including less dependence on generic review and Regulatory Guides, and greater accessibility to documentation.

- Of importance is a transition from the legalistic emphasis currently employed to an objective technical assessment.

- Finally, top priority should be given to handling the issues of documentation of deviations, assessment of cumulative impact of deficiencies, and reversal of the current backfit procedure so as to insure proper attention is given to safe operation of plants.

- OTHER ISSUES (SECTION 4.7)

Two other broad issues of significance were identified.

Programs should be initiated to:

- Standardize the periodic review of operating plants and LO

-Achieve independent status of those individuals serving on the ACRS, the ASLB and the plant Review Boards.

6.2: PRIORITIES

The subject of Program Priority Evaluation is discussed extensively in Section 5. In general, it is concluded that a cost/benefit type analysis is the logical choice for program priority setting. The drawback to this technique is, however, that nuclear safety evaluation uncertainties are of such a magnitude that direct quantification cannot yet be used as the primary selection process. Quantification should be performed where possible, but the final choices continue to be matters of informed judgment.

The review of the study programs identified in Sections 2, 3 and 4 and further summarized in Section 6.1, has accordingly not included an attempt at specific quantification or exact ranking in order of priority. It is understood that major decision process studies are under way at other DOE contractors, and it is recommended that the methodologies developed be applied to ranking the programs recommended in this report. It is also recommended that further efforts be made to, for example, determine which plants contribute the largest to the nuclear program risk, through means of evaluating applicable characteristics

contained in Table 5-1. Such quantification would serve to reduce the uncertainty.

It is understood that part of the value of this Study to Sandia was to obtain the ranking of the recommended programs based on our experience and informed judgment. Accordingly, the following advice is offered, with the understanding that these principles apply primarily to decision on programs to be funded by the federal government.

- -All of the programs and recommendations contained in this report are considered to be of substantial import. Low priority tasks have already been eliminated.
- -All programs which serve to make the design, regulatory, inspection and other process functions more effective should be given the highest priority. It will serve no purpose to invent ultra-safe systems if they continue to be valved out of service at critical times.
- -Programs which are directly applicable to existing operating plants should receive top priority. There is some reason to believe that improvements have been made over the past years, and the older plants are believed to be the greatest contributors to risk.
- -Finally, future federal programs should focus more on consequence mitigation than on probability reduction. The past belief that accidents will not happen has encouraged

much of the past effort to be applied to the probability reduction side of the ledger. Mitigation should get more attention. This position is further justified by the belief that the utilities and suppliers should be funding the bulk of the probability improvement work.

With these principles in mind, it is our belief that the programs outlined in this report should be ranked as follows:

Safety Assurance Programs (Section 4)	-	Highest
Consequence Mitigation (Section 3)	-	Medium
Probability Reduction (Section 2)	-	Lower

It should again be emphasized that all programs listed are needed, and the ranking here is presented only for purposes of allocation of federal funding. Recognizing that even federal funding is subject to budgeting limits, we further recommend that priority ranking be set by the two-level screening process depicted in Figure 5-1. This will ensure that all items are addressed.

6.3: FUTURE WORK

As in any study conducted in a limited time frame, this study has of necessity been limited in its evaluation of both the number and the significance of potential safety improvement issues. It is believed, however, that the work presented here is of relatively broad scope, providing an overview of the more significant issues affecting reactor safety. It should be emphasized, therefore, that each of the issues and associated recommendations

can and should be examined in more depth before extensive programs are initiated. This will undoubtedly occur in the budget review process, but a special effort should be made to ensure that an objective assessment is made periodically to verify that the safety goals being sought are still valid.

Re-examination of the program in six to twelve months seems of particular relevance at this time because of the uncertainty of the forthcoming post-TMI review recommendations. While there is need to respond to the reality of the TMI accident, there is also a danger that the safety program may "react" exclusively to it. The safety improvement program should be subjected to an ongoing or periodic review to guard against this possibility.

It is particularly encouraging to see the steps now being proposed to establish a program for the independent analysis of operational data.^{1/} This program, proposed to be a new office reporting to the NRC Director For Operations should help to achieve some of the regulatory effectiveness discussed in Section 4. While it is too early to predict or assess how this organization may work, it should be an obvious focal point for the identification of future research efforts. We strongly recommend that all possible support be provided to ensure it becomes an effective and truly independent part of the safety assurance structure.

SECTION 6

REFERENCES

- 1/ Inside NRC, August 27, 1979, page 7, McGraw-Hill publication.

LIST OF APPENDICES

APPENDIX

DESCRIPTION

A	Minutes-Public Meeting of President's Commission on the Accident at Three Mile Island, Unit 2.
B	Minutes-NRC Briefing of Operating Plant Owners and Owners with Near Term OL's.
C	Viewgraphs-Presentations to ACRS Subcommittee on Improved Safety Systems.
D	Viewgraphs-Presentation of R. Mattson to NRC Commissioners on TMI-2 Lessons Learned Task Force.
E	Recommendations of the Nuclear Power Plant Emergency Review Panel.

APPENDIX A

MINUTES

PUBLIC MEETING OF PRESIDENT'S COMMISSION ON THE

ACCIDENT AT THREE MILE ISLAND, UNIT 2

WASHINGTON, D. C.

AUGUST 21, 22, 23, 1979

TECHNICAL ASSOCIATES

TECHNICAL CONSULTANTS ON ENERGY & THE ENVIRONMENT

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September 1, 1979

Dr. Arthur DuCharme
SANDIA LABORATORIES
Org. 4443
Box 5800
Albuquerque, NM 87185

SUBJECT: Trip Report - President's Commission on TMI
Public Hearing August 21-23, 1979

Dear Art:

The final public hearings for the President's Commission* on the accident at TMI were held at Georgetown University in Washington, DC on August 21-23, 1979. In connection with our study of possible ways to improve LWR safety, I attended these hearings and made the following observations related to LWR safety:

- I Governor Thornburgh (PA) clearly conveyed the difficulty he experienced in attempting to deal with numerous agencies, unknown individuals (even from the NRC), faulty information, unreliable sources, and crank calls. From this input data, he was attempting to make serious and critical decisions on evacuation measures. It is clear the safety of an LWR during an accident is very difficult to discern--particularly from outside the plant facility. This area deserves more attention in future research by DOE.

A strong recommendation which came out of Gov. Thornburgh was to echo Denton's comment that you can't manage control of an accident from Washington, DC; it requires competent people at the site and good communications. He also recommended the following:

- Siting in the future should be more rigorously reviewed.
- Siting decisions should be expanded to include state and other bodies in the siting process.
- States should have the right to visit, monitor, and inspect its local nuclear sites and facilities.

- States need to be better prepared to handle these types of accidents.
- In his view, siting can no longer be left to the experts; we must seek local citizens' input.

Regarding his evacuation decisions, he was concerned for the following:

- There is risk in the process of evacuation - elderly people, ICU patients, babies, traffic.
- Evacuations of this nature had never been carried out anywhere in the world.
- A 5 mile diameter evacuation has ramifications out to 100 miles and beyond.

II Joseph LaFleur, NRC International Programs, was questioned at length about the feedback system on safety problems. They were particularly concerned about an accident in a Westinghouse plant in Beznau, Switzerland which could have been considered a precursor to TMI. Westinghouse used a coincidence logic on HPI (low pressure and low-level to initiate) at Beznau as they did in the U.S. The accident on September 4, 1974 followed a sequence close to TMI.

Turbine trip.

Primary pressure rose

PORV failed to close

Pressurizer level rose

No HPI until 11½ minutes (low pressure & low level)

Block valve in PORV closed to stop boiling

Lots of water lost; ruptured quench tank seal

Operator confusion was reported as occurring in the Swiss accident also. The Beznau event wasn't reported to NRC until April 1979. LaFleur didn't know about it until the President's Commission told him about it.

This event points up the need for research into more effective accident reporting and information evaluation techniques to discern the impact on safety of other plants.

III James Cresswell (NRC inspector), Jesse Ebersole (ACRS), Paul Collins (operator training), Roger Matson (NRC, DSS), Harold Denton (NRC, NRR), Victor Stello (NRC, Director I&E), and Tony Roisman (NRDC) then paraded through the witness stand. The thrust of the questioning had to do with the processes by which safety issues were raised and handled within the NRC. There were several examples where the process failed to expose or excessively delayed exposure of safety concerns.

Among the safety issues mentioned are the following:

1. Reactor vessel level measurement: This issue has been around since 1974. According to Ebersole, the ACRS has questioned the reliance on pressurizer level to infer a vessel level. The vendors view (again according to Ebersole) was that they "don't need reactor vessel level instruments; don't even need anything if they would just let the equipment do its job."

This is worthy of research into the needs for automation versus operator action and under what conditions can the operator rely totally on the equipment (i.e., the 10 minute rule). How tolerant is the system to misaligned valves, etc.?

2. Natural circulation problems: These are caused by steambinding or vapor lock. These problems were brought out on Westinghouse and Combustion Engineering plants including Diablo Canyon. It is still an unresolved issue.
3. The ACRS's right to raise new safety issues was described as being limited. They are to review the specific design, not make suggestions for changes to design. The same problem exists for utilities in that they are reluctant to raise safety questions; instead they wait for regulatory action.
4. Operator training and testing has some serious limitations in that the tests may be available to some applicants from previous operators, the

walk-through portion does not adequately test accident mitigation knowledge and the simulators are not always representative of the plant and control room to be operated.

5. Roger Matson confirmed that no one is in charge of reviewing the control room design and human factors aspects of the man-machine interface. Several accidents have been influenced by the man-machine interface and the area is ripe for research.
6. Once-through-steam-generator designs and the B&W reactors were characterized by Matson as being "less forgiving." The question of how far one can go in this direction before it is classed as potentially unsafe deserves some additional study.
7. Sabotage was discussed briefly in the hearings. One of TMI's operators said an operator could cause a core melt down from the control room. This caused the President's Commission to question whether current regulations are sufficient to protect from insiders as well as outsiders. They also discussed the need for developing better psychological criteria for operator selection.
8. Station blackout can cause a loss of all feedwater at some plants.
9. Extended period small LOCA's are not adequately evaluated in present licensing processes.

IV The final day of the hearing was consumed with the President's Commission's outrage over Harold Denton's decision to resume licensing processes. Denton's memo (copy attached) was treated as an affront to the Commission's effort and a liberal amount of reprimand fell on Denton and Commissioner Kennedy.

September 1, 1979

These were the final public hearings of the Kemeny Commission and based on these hearings alone, the Commission appears prepared to make some very sweeping and strong recommendations in their final report due October 25, 1979:

Sincerely,

MHB TECHNICAL ASSOCIATES

Gregory C. Minor

GCM:kc
Enclosure

* A list of the Commissioners is attached.

PRESIDENT'S COMMISSION ON THE
ACCIDENT AT THREE MILE ISLAND

John G. Kemeny, Chairman
President of Dartmouth College

Bruce Babbitt
Governor of Arizona

Patrick E. Haggerty
Retired President of Texas Instruments

Carolyn Lewis
Associate Professor of Journalism
Graduate School of Journalism,
Columbia University

Paul E. Marks
Vice President for Health Sciences, Columbia University

Cora B. Marrett
Associate Professor of Sociology at the University of Wisconsin

Lloyd McBride
President of United Steelworkers of America

Harry McPherson
Attorney

Russell Peterson
President of Audubon Society

Thomas Pigford
Professor and Chairman of Department of Nuclear Engineering
at the University of California at Berkeley

Theodore Taylor
Professor of Aerospace and Mechanical Science at
Princeton University

Anne Trunk
Resident of Middletown, Pennsylvania

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

file copy

August 20, 1979

MEMORANDUM FOR: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Bradford
Commissioner Ahearne

THRU: Lee V. Gossick
Executive Director for Operations *[Signature]*

FROM: Harold R. Denton, Director
Office of Nuclear Reactor Regulation

SUBJECT: RESUMPTION OF LICENSING REVIEWS FOR NUCLEAR POWER PLANTS

In May of this year I described a realignment of current and near-term priority tasks within the Office of Nuclear Reactor Regulation (NRR) to deal with activities relating to the accident at Three Mile Island (see SECY-79-344). One consequence of the realignment was a temporary delay in the processing of operating license and construction permit applications for nuclear plants pending completion of certain TMI-2 related tasks.

The short-term TMI-2 tasks are essentially complete, as summarized below, and based on the results of these efforts I have decided to resume staff licensing activities on pending construction permit and operating license applications. It is my judgment that the TMI-2 related actions being taken by NRR on licensee emergency preparedness (see SECY-79-450), operator licensing (see SECY-79-33-E), bulletins and orders followup (primarily in the areas of auxiliary feedwater system reliability; loss of feedwater and small break loss-of-coolant accident analysis; emergency operating guidelines and procedures; and operator training), and short-term Lessons Learned, if accomplished generally on the schedule we have selected, are necessary and sufficient for the continued safe operation of operating plants and for the resumption of staff licensing activities on pending construction permit and operating license applications. It is my intent to bring the staff's first completed review of a pending operating license application to the Commission for review prior to staff issuance of the license. The Lessons Learned Task Force and I also have considered whether the actions associated with these activities would foreclose other actions that subsequently may be shown to be necessary by the Lessons Learned Task Force, the President's Commission or the NRC Special Inquiry. We have no indication that they will.

The principal element of the composite of staff activities listed above is the completion of my review and the ACRS review of the first report of the TMI-2 Lessons Learned Task Force (NUREG-0578). The Task Force report contains a set of recommendations to be implemented in two stages over the next 16 months on operating plants, plants under construction, and pending construction permit applications. The Task Force recommended 20 licensing requirements and three rulemaking matters in 12 broad areas (nine in the area of design and analysis and three in the area of operations). All but one of the 23 recommendations had a majority concurrence by the Task Force. The Task Force concluded that implementing its recommendations would provide substantial, additional protection which is required for the public health and safety.

The Advisory Committee on Reactor Safeguards has completed its review of the Task Force report. The several public meetings of the ACRS subcommittee on TMI-2 and the public meeting of the full committee on August 9 provided an opportunity for the presentation and discussion of public comments on the report. The ACRS letter of August 13, 1979, to Chairman Hendrie states that the Committee agrees with the intent and substance of all the Task Force recommendations, except four upon which the Committee offered constructive comments to achieve the same objectives articulated by the Task Force. The Committee also noted that effective implementation will require a more flexible, perhaps extended, schedule than proposed by the Task Force. A copy of the ACRS letter is provided as Enclosure 1.

The ACRS comments on NUREG-0578 concentrate on four of the Task Force recommendations. These are: (a) the revision of limiting conditions of operation to require plant shutdown for certain human or procedural errors; (b) the inerting of MKI and II BWR containments; (c) the provision of recombiner capability at operating plants that do not already have it; and (d) the addition of a shift technical advisor at each operating plant. The first three of these matters require Commission rulemaking, and it is a straightforward matter for the staff to consider the comments in the process of developing the required Commission papers. I will assure that is done.

It is my intent to ask the Office of Standards Development (SD) to proceed expeditiously with a Commission paper proposing a new rule on limiting conditions of operation (item a, above). I will ask SD to include in the paper the alternative approach recommended by the ACRS, and one other approach that I think merits consideration. My alternative would amend the Task Force recommendation so as to differentiate between an isolated occurrence and a repetitive pattern. For example, the forced shutdown aspect of the Task Force recommendation could be reserved for a repeat violation within a relatively short time period, such as two years.

In the case of the two hydrogen control matters (items b and c, above), I intend to follow the advice of the ACRS by asking SD to delay completion of the required staff papers for proposed rulemaking until after receipt and review of the final report of the Lessons Learned Task Force, now scheduled for completion in mid-September. It is likely that the inerting and recombiner requirements recommended by the Task Force will be included in the eventual solution to the hydrogen control problems encountered in the TMI-2 accident. However, in view of the short time until the availability of the overall hydrogen control recommendations by the Task Force, I agree with the ACRS that it is best to not dilute staff effort in this area by prompt pursuit of the two short-term recommendations, one of which was a minority view of the Task Force for these same reasons.

The ACRS comments on the shift technical advisor (item d, above) have resulted in our reassessment of the possible means of achieving the two functions which the Task Force intended to provide by this requirement. The two functions are accident assessment and operating experience assessment by people onsite with engineering competence and certain other characteristics. I agree with the Task Force that the shift technical advisor concept is the preferable short-term method of supplying these functions. However, I have concluded that some flexibility in implementation may yield the desired results if there is management innovation by individual licensees. The Task Force has prepared a statement of functional characteristics for the shift technical advisor that will be used by the staff in the review of any alternatives proposed by licensees. It is provided here as Enclosure 2.

In addition to commenting on four of the Task Force recommendations, the ACRS letter of August 13 recommends three additional instrumentation requirements for short-term action. These are containment pressure, containment water level and containment hydrogen monitors designed to follow the course of an accident. I agree with these recommendations. The Task Force has prepared descriptions of these requirements in the same format as Appendix A of NUREG-0578. They are provided here in Enclosure 3.

I have also decided on one further licensing requirement for short-term action. It is a requirement for remotely operable high point venting of gas from the reactor coolant system. The Task Force has prepared a description of this requirement; it is provided here in Enclosure 4. The Task Force had previously deferred this item for further study, but it is my judgment that design efforts by licensees can and should be initiated now.

Finally, the Task Force has compiled a set of errata and clarifying comments for NUREG-0578. It is provided here as Enclosure 5.

The Commission

-4-

In summary, the Task Force recommended prompt licensing action on 20 items (excluding the three rulemaking matters). I have added the three additional requirements recommended by the ACRS in its August 13 letter and one more on the basis of my own review. This Office will issue letters to all operating plant licensees and all construction permit and operating license applicants within the next two weeks requiring them to commit within 30 days to meet the total of 24 licensing requirements on the implementation schedule provided here in Enclosure 6. Another letter to be issued at approximately the same time, will state the requirements flowing from the work by the Bulletins and Orders Task Force on operating plants which also need to be picked up on the license applications:

Several licensees have advised that some of the hardware changes required in NUREG-0578 can be accomplished at much lower cost during springtime refueling outages in 1980. For good cause shown, we intend to consider such flexibility in the implementation schedules. The end date for full implementation of all licensing requirements has not been changed from the January 1, 1981, date recommended by the Task Force. The implementation dates for the Commission rulemaking actions will be established in the course of rulemaking.

Harold R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:

1. ACRS Ltr Carbon to
Hendrie dtd 8/13/79
2. Alternatives to Shift Technical
Advisors
3. Instrumentation to Monitor Containment
Conditions
4. Installation of Remotely Operated High Point
Vents in the Reactor Coolant System
5. NUREG-0578 Errata
6. Implementation of Requirements for Operating
Plants and Plants in OL Review

Enclosures 1 to 6
are not included
in Appendix A.

cc: Mitchell Rogovin
Saul Levine
Robert Minoque
Victor Stello
William Dircks
Carlton Kammerer
ACRS

APPENDIX B

MINUTES

NRC BRIEFING OF OPERATING PLANT OWNERS

AND OWNERS WITH NEAR TERM OL'S

WASHINGTON, D. C.

AUGUST 1, 1979

NRC BRIEFING - OL/OPERATING PLANTS (8/1/79)

- A. Purpose of the meeting was for the NRC to brief the utilities with operating plants and near terms OL's of the present recommendations of the following 4 task forces:

1. Lessons Learned Task Force (LLTF) - Roger Mattson
2. Bulletins and Orders - Denny Ross
3. Emergency Response Plans - Bryan Grimes
4. Operator Training - Paul Collins

B. Denton:

1. Will report on progress re analysis of TMI accident. Will inform re what about "to lay on you." Reviewing again if NRC Commissioners approve the 4 items (CP's and OL's).
2. Suggests new ways to handle amendments to OL's. All utilities should agree re alternatives - one report for 72 reactors - list alternatives. Wants to break out of case-by-case/docket-by-docket method of license amendment reviews.

C. R. Mattson (See NUREG-0578 for details):

1. Resumption of licensing - necessary and sufficient response if meets LLTF's 23 recommendations in 12 categories. Implement in two stages; Jan. 1, 1980 and Jan. 1, 1981. (Some too short; some too long.) Will allow, if good cause is shown, to lengthen or shorten the schedule. Can't tell difference between 5 and 8 months. Would like completion before Jan. 1, 1981.
2. Like Denton, favors a lead plant concept. Likes Salem-2 and North Anna-2 as lead plants for OL's.
3. Near-term OL's - after long-term report in Sep., LLTF will write SER Suppl. for TMI changes. Will review the implementation on a plant-by-plant basis.
4. Design Analysis Recommendations
 - a) (LLTF 2.1.1): Only PWR's. Frequency is "1" vs "10⁻³ to 10⁻⁴" for use of HPCI. Therefore, must improve reliability of PORV. NRC Probability Assessment Branch has done unreliability analysis of (W) and (CE) feedwater systems. Most new plants meet this requirement.

- b) (LLTF 2.1.2): Get on with full-scale quality testing - involved in ATWS review. Ability of valves to reclose. Quality testing for off-normal conditions - many logical extensions of this task. Quality testing to be completed by July 1, 1981. Three approaches accepted by the NRC:
1. Germany, France, and Japan have facilities under construction.
 2. Wylie Labs - existing but undersized facility.
 3. GE re MSIV testing - obtained a facility like Staveline.
- NRC may be a third party for such a test. Test plans to NRC by Jan. 1980.
- c) (LLTF 2.1.3):
1. Close to direct releasing instrumentation.
 2. Detection re inadequate core cooling.
 3. Direct measurement of vessel water level.
- Also human engineering re plant diagnostics - lessons for the future.
- d) (LLTF 2.1.4):
1. Implement SRP position. May, in future, want to isolate on radiation (not presently done).
 2. Non-safety systems used other than expected. Used in a safety manner (reviewing essential/non-essential categories).
 3. Automatic reset of containment isolation valves upon reset.
- e) (LLTF 2.1.5): Inert Mark I and Mark II. Fifty (50) plants with no recombiner capability. TMI had 30-50% metal water reaction. Staff must answer the 30-50% vs 5% in regulations. Large PWR's can withstand TMI-type explosions better than BWR's. PWR's containment is 2,000,000 vs Mark I & II BWR's of 300,000 cu. ft. Also, single failure proneness of valves.

- f) (LLTF 2.1.6): Safety systems used in methods never intended. Particularly the "let down system." Need for qualification and reliability extends beyond safety systems. A continuum of sorting to be developed - example: earthquakes. In interim to call for leakage testing and periodic testing program. Also a shielding review for safety & non-safety systems. Three aspects - add shielding or replace equipment. TMI safety systems couldn't cope with TMI radiation levels (assume TID source term change).
- g) (LLTF 2.1.7): Applies only to PWR's - automatic initiation.
- h) (LLTF 2.1.8): Post-accident monitoring is a 10-year controversy - some warranted and some not.
 - 1. Approx 20 instruments within state of art. Should have been put on years ago. Developing a revision of Reg Guide 1.97 by Sep. Week to 2-week long drafting session.
 - 2. Beyond the state of art - resolve difficulties of the research.
 - 3. Discern iodine from other noble gases.
- i) (LLTF 2.1.9): All analysis items. Three areas include small break analysis, core uncovering instruments, (gaming analysis for upset conditions), single failures and joint event trees and sequence trees.

5. Operations

- a) (LLTF 2.2.1):
 - 1. Control Room Commander - the Senior Reactor Operator. Put in decision-making role - not 70% of time on administrative matter. Establish clear lines of control & authority.
 - 2. Shift turnover procedures - formal & precise. Also for technicians, etc.
 - 3. Shift technical advisor - wants on-site (that's the argument) study LER's, procedures, etc. Devoted to safety. Couldn't convince self (NRC) that 30 minutes or 5 minutes away is acceptable, (need immediate diagnosis during upset conditions). Want by Jan. 1, 1980. Very strong on this recommendation. Much disagreement re "qualifications." Denton - issue is accountability. Could beef up SRO vs technical advisor.

(Mattson underlined need for better man-machine interface & diagnostic aids. Will take a couple of years. Didn't address in (LLTF). Could upgrade the SRO.)

- b) (LLTF 2.2.2): Three parts - limited control room access, onsite technical support center, onsite operations support center (no habitation requirements).
- c) (LLTF 2.2.3):
 - 1. A rule-making proposal. 3,000 LER's per year (1% which result in total loss of safety function). One approach was more NRC involvement. Then, shutdown vs a fine. 139 loss of safety findings of 2,640 occurrences in 10 years of operation due to human or operational error. Some utilities with no occurrences - others with many.
 - 2. (CP & L) - additional pressures - raise rates to consumers and wants to give input prior to
 - 3. (?) - What about actions of a disgruntled employee?

6. General

- a) Kenon - VT Yankee: Can we really meet Jan. 1, 1980 for Category A items. Example of Mark I. Mattson believes Category A are primarily procedural-administrative. Could revise equipment at next regular shutdown.
- b) Denton - is thinking of adding automatic reactor head venting system to LLTF recommendations.

D. Bulletins - Orders: Denny Ross (See Attachment 1 for details)

- 1. Last bulletin issued last week (79-05C&06C). Also intend to issue no more orders. Still to complete look at small LOCA and loss of feedwater event.
- 2. (page 2) Evaluation sequence.
 - a) Jul. 9 - ACRS Subcommittee meeting.

3. Long-term B&W orders

- a) FM&EA of integrated control system.
- b) Safety grade reactor trip in loss of main feedwater.
- c) Improvements in auxiliary feedwater systems
- d) Continued operator training.

4. (W) plants proposed schedule and chronology included.

5. Listed 15 items applicable to BWR's.

6. Near-term requirements for (W) plants.

- a) Auxiliary feedwater system - 9 items
- b) Plant-specific - 4 items.
Diverse feedwater (Davis-Besse). Electric as well as turbine-driven feedwater.
- c) Procedures - 4 items.

7. Whatever B&O requires on Salem-1 and North Anna-1 will be required on Salem-2 and North Anna-3 prior to fuel loading.

8. Denton has obtained 50-60 man-years from other government agencies for use in licensing. The borrowed employees will not be used on the next 7 OL's.

E. Bryan Grimes - Emergency Planning for Reactor Owner (See Attachment 2)

- 1. Regional meetings near the end of August to brief the utilities (August 20 for Diablo).
- 2. Elements of Staff effort:
 - a) Upgrade to Reg Guide 1.101.
- 3. Denton - feels that the AIF plan (Warren Owens) looks good. Should be off-site center for political/top management info (also instruments possibly).
- 4. NRC will recommend that FEMA fund some of the state's work.

F. Paul Collins - Operators (See Attachment 3)

1. Sent 23 recommendations to NRC Commissioners - 15 recommended for approval.
2. Recommendations 1 to 6 address operator training.
 - a) Denton wants multiple failure accidents at the simulator for "off-normal" conditions.
3. Recommended 7, 8, & 9 re requalification program.
4. Recommended 10 & 13 re NRC written examinations.
5. Recommended 11, 12, & 15 re NRC operating tests.
6. Recommended 14 is same as 13 actually.
7. With the new pass/fail rules on operator exams, approximately 50% of the reactor operators and senior reactor operators would have failed the examinations.

G. Darrell Eisenhut

1. Additional letter to each licensee on LLTF following Denton & Commissioners' review.
2. In mid-August - B & O letters will be sent out.
3. Emergency preparedness letters sent out today re regional meetings.

H. Vassallo re Near-term OL Reviews

1. Will send out formal letters on LLFT, B & O, and emergency preparedness. Will resolve prior to an OL issuance.
2. Also working on other relevant issues.

I. Denton

1. Plea for Jan. 1, 1981 utility completion of all TMI-2 related recommendations. Wants to be able to report the preceding to Congress, public, etc.

BULLETINS & ORDERS TASK FORCE

PURPOSE

- o REVIEW GENERIC IMPLICATIONS OF TMI-2 ACCIDENT FOR ALL OPERATING PLANTS TO CONFIRM BASES FOR THEIR CONTINUED SAFE OPERATION.
- o ADVISE LESSONS LEARNED TASK FORCE OF ANY ACTIONS IDENTIFIED DURING THE REVIEW.

SCOPE OF REVIEW

- o LOSS OF FEEDWATER EVENT
 - . ANALYSIS
 - . SYSTEMS
 - . GUIDELINES AND PROCEDURES
 - . OPERATOR TRAINING
- o SMALL BREAK LOCA
 - . ANALYSIS
 - . SYSTEMS
 - . GUIDELINES AND PROCEDURES
 - . OPERATOR TRAINING

B&O EVALUATION SEQUENCE

BULLETINS / ORDERS

D. ROSS

T. NOVAK (DEPUTY)

PROJECTS
(HELTEMES)

ANALYSIS
(ROSZTOCZY)

SYSTEMS
(ISRAEL)

- C-E PLANTS (M. VILLALVA)
- H PLANTS (P. O'REILLY)
- B&W PLANTS (R. CAPRA)
- GE PLANTS (W. YAME)
- ORPM (AS NEEDED)

- RSB
- ASB
- PSB
- OLB
- CSB
- ICSB

ACRS SUBCOMMITTEE
ON
BULLETINS AND ORDERS

- FORMED IN MID-JUNE 1979
- MEMBERSHIP

W. MATHIS (CHAIRMAN)

M. BENDER

H. ETHERINGTON

S. LAWROSKI

M. PLESSET

P. SHEWMON

B&O REVIEW MATTERS

B&W PLANTS

- o LONG-TERM ACTIONS IDENTIFIED IN COMMISSION ORDERS FOR B&W PLANTS
- o LONG-TERM ACTIONS IDENTIFIED IN STAFF SER'S ON B&W PLANTS
- o NEAR-TERM ACTIONS IDENTIFIED IN NUREG-0578 RELATING TO B&O
REVIEW SCOPE

REVIEW MATTERS IDENTIFIED IN COMMISSION ORDERS
AND STAFF SER'S FOR B&W PLANTS

COMMISSION ORDERS

- o FAILURE MODES AND EFFECTS ANALYSIS OF INTEGRATED CONTROL SYSTEM
- o SAFETY-GRADE REACTOR TRIP IN LOSS OF MAIN FEEDWATER
- o IMPROVEMENTS IN AUXILIARY FEEDWATER SYSTEMS
- o CONTINUED OPERATOR TRAINING

STAFF SER'S

- o TRANSIENT ANALYSES
- o SMALL BREAK LOCA ANALYSES
- o SAFETY AND RELIEF VALVE STUDIES
- o PRESSURE VESSEL INTEGRITY
- o AUXILIARY FEEDWATER SYSTEM RELIABILITY STUDY
- o AUXILIARY FEEDWATER SYSTEM CONTROL TESTS

SCHEDULE AND CHRONOLOGY OF GENERIC REVIEW
OF B&W PLANTS

<u>ACTIVITY</u>	<u>DATE</u>
ISSUE ORDERS	MAY 7-17, 1979
ISSUE SER; LIFT ORDERS	MAY 18 - JULY 6, 1979
<hr/>	
LICENSEES SUBMITTAL OF SCHEDULE FOR LONG-TERM ACTIONS	JUNE 18 - MID AUGUST 1979
REVIEW LONG-TERM ACTION; ISSUE EVALUATION	ONGOING
EVALUATE ACCIDENTS AND TRANSIENTS BEYOND CURRENT DESIGN BASES	DECEMBER

SCHEDULE AND CHRONOLOGY OF GENERIC ASSESSMENT
OF WESTINGHOUSE-DESIGNED OPERATING PLANTS

INITIATE GENERIC REVIEW	MAY 1, 1979
MEETINGS WITH LICENSEES ON AUXILIARY FEEDWATER SYSTEMS	MAY 22-26, 1979
MEETINGS WITH LICENSEES REGARDING FORMATION OF OWNERS' GROUP	MAY 30, 1979
MEETING WITH SMALL BREAK ANALYSIS SUBCOMMITTEE OF OWNERS' GROUP	MAY 31, 1979
GENERIC REQUESTS FOR INFORMATION ISSUED	JUNE 4, 1979
MEETING WITH PROCEDURES SUBCOMMITTEE OF OWNERS' GROUP	JULY 18, 1979
<hr/>	
ISSUE STAFF INSTRUCTIONS TO LICENSEES	MID-AUGUST 1979
ISSUE GENERIC REPORT	EARLY-SEPTEMBER 1979
LICENSEE RESPONSES TO INSTRUCTIONS	MID-SEPTEMBER 1979
ISSUE STAFF EVALUATION ON INITIAL PLANT	MID-OCTOBER 1979
EVALUATION OF ACCIDENTS & TRANSIENTS BEYOND CURRENT DESIGN	DECEMBER

SCHEDULE AND CHRONOLOGY OF GENERIC ASSESSEMENT OF
COMBUSTION ENGINEERING DESIGNED OPERATING PLANTS

<u>EVENT</u>	<u>DATE (1979)</u>
INITIATE GENERIC REVIEW OF C-E DESIGNED OPERATING PLANTS	MAY 1
MEET WITH LICENSEES REGARDING AUXILIARY FEEDWATER SYSTEMS	MAY 22 - 25
MEET WITH LICENSEES AND C-E REGARDING THE FORMING OF A C-E OWNER'S GROUP	JUNE 12
MEET WITH ANALYSIS SUBCOMMITTEE OF OWNER'S GROUP	JUNE 15 & JULY 24
MEET WITH PROCEDURES AND GUIDELINE SUBCOMMITTEE OF OWNER'S GROUP	JUNE 29 & AUGUST 10
ISSUE STAFF REQUIREMENTS TO LICENSEES	LATE AUGUST
ISSUE GENERIC REPORT	MID SEPTEMBER
APPLICANTS RESPOND TO STAFF REQUIREMENTS	LATE SEPTEMBER
ISSUE STAFF EVALUATION REPORT ON INDIVIDUAL PLANTS	LATE OCTOBER (FIRST REPORT)
EVALUATION OF ACCIDENTS & TRANSIENTS BEYOND CURRENT DESIGN	DECEMBER

B&O REVIEW MATTERS

BOILING WATER REACTOR PLANTS

- o NUREG-0578 NEAR-TERM REQUIREMENTS RELATING TO B&O REVIEW SCOPE
- o NEAR-TERM REQUIREMENTS RESULTING FROM B&O GENERIC REVIEW OF BWR'S

PROSPECTIVE NEAR-TERM REQUIREMENTS

B&O GENERIC REVIEW OF

BOILING WATER REACTOR PLANTS

1. EXTEND RANGE OF WATER LEVEL RECORDERS IN THE CONTROL ROOM; INITIATE EXTENDED RANGE RECORDERS ON REACTOR TRIP.
2. AD'S INITIATION ON EITHER (LOW)³ WATER LEVEL OR HIGH DRYWELL PRESSURE (IN CONJUNCTION WITH OTHER PERMISSIVES).
3. PROVIDE FOR AUTOMATIC RE-INITIATION OF RCIC AND HPCI ON (LOW)² LEVEL.
4. PROVIDE FOR ISOLATION OF VENTING FROM ISOLATION CONDENSERS ON HIGH RADIATION LEVELS.
5. REDUCE FAILURE OF RELIEF VALVES BY REDUCING CHALLENGE RATE AND/OR IMPROVING DESIGN.
6. MODIFY OPERATING PROCEDURES TO REQUIRE A LOW PRESSURE SYSTEM RUNNING (LPCI, CORE SPRAY, CONDENSATE SYSTEM) BEFORE MANUAL DEPRESSURIZATION.
7. MAKE PROCEDURES CONSISTENT WITH OPERATOR SCENARIOS BEING DEVELOPED.
8. DEVELOP GENERIC GUIDELINES FOR EMERGENCY PROCEDURES.
9. ADDITIONAL OPERATOR TRAINING.
10. ESTABLISH AND MAINTAIN ONE SET OF AS-BUILT DRAWINGS ON SITE.
11. DETERMINE ROLE OF RECIRCULATION PUMPS IN CASE OF INADEQUATE CORE COOLING.
12. INVESTIGATE NATURAL CIRCULATION WITH CORE SPRAY.
13. ANALYSIS OF BREAK IN RECIRCULATION LINE TO DETERMINE WHETHER ISOLATION VALVES SHOULD BE CLOSED.
14. DEVELOP IMPROVED SMALL BREAK METHODS.
15. VERIFY BY EXPERIMENT THE SMALL BREAK LOCA METHODS.

SCHEDULE AND CHRONOLOGY OF GENERIC
REVIEW OF BWR'S

<u>ACTIVITY</u>	<u>DATE</u>
INITIATE GENERIC REVIEW	JUNE 7, 1979
MEETING WITH LICENSEES	JUNE 28, 1979
GENERIC REQUESTS FOR INFORMATION ISSUED	JULY 13, 1979
<hr/>	
ISSUE GENERIC REPORT AND INSTRUCTIONS TO LICENSEES	MID- SEPTEMBER
LICENSEE RESPONSES TO INSTRUCTIONS	MID-OCTOBER
INITIAL STAFF EVALUATION ON INDIVIDUAL PLANTS	MID-NOVEMBER
EVALUATE ACCIDENTS AND TRANSIENTS BEYOND CURRENT DESIGN BASES	DECEMBER

NEAR TERM REQUIREMENTS

FOR WESTINGHOUSE AND COMBUSTION ENGINEERING OPERATING PLANTS

AUXILIARY FEEDWATER SYSTEM

GENERIC

- . AUTOMATE (LL)
- . SINGLE SUCTION VALVE
- . ALTERNATE WATER SOURCE
- . CST LOW LEVEL ALARM
- . PUMP ENDURANCE TEST
- . FLOW INDICATION (LL)
- . VALVE POSITION (LOSS OF AIR)
- . ACTUATION ON LOSS OF ALL AC
- . TECH SPECS

o PLANT SPECIFIC

APW PUMP TEST CRITERIA

MODIFY VALVE LINEUPS

REVIEW COMMON MODE ELECTRICAL FAILURES

MODIFY SURVEILLANCE TEST PROCEDURES

PROCEDURES

- o SMALL LOCA
- o EXTENDED LOSS OF FEEDWATER
- o SG DUMP VALVE OPERATION
- o TRAINING

ANALYSIS

- o SMALL BREAK LOCA
- o EXTENDED LOSS OF FEEDWATER
- o MICHAELSON CONCERNS
- o VENDOR GUIDELINES
- o CODE VERIFICATION
- o INADEQUATE CORE COOLING SYMPTOMS/OPERATOR ACTIONS
- o STAFF SMALL BREAK AUDITS

July 23, 1979SECY-79-450

For: The Commissioners

Thru: Executive Director for Operations *TAR for L.V.G*

From: Harold R. Denton, Director, Office of Nuclear Reactor Regulation

Subject: ACTION PLAN FOR PROMPTLY IMPROVING EMERGENCY PREPAREDNESS

Purpose: To inform the Commission of the staff's plans to take immediate steps to improve licensee preparedness at all operating power plants and for near-term OL's.

Discussion: While the emergency plans of all power reactor licensees have been reviewed by the staff in the past for conformance to the general provisions of Appendix E to 10 CFR Part 50, the most recent guidance on emergency planning, primarily that given in Regulatory Guide 1.101 "Emergency Planning for Nuclear Power Plants", has not yet been fully implemented by most reactor licensees. Further, there are some additional areas where improvements in emergency planning have been highlighted as particularly significant by the Three Mile Island accident.

The NRR staff plans to undertake an intensive effort over about the next year to improve licensee preparedness at all operating power reactors and those reactors scheduled for an operating license decision within the next year. This effort will be closely coordinated with a similar effort by the Office of State Programs to improve State and local response plans through the concurrence process and Office of Inspection and Enforcement efforts to verify proper implementation of licensee emergency preparedness activities.

The main elements of the staff effort, as listed in Enclosure 1, are as follows:

- (1) Upgrade licensee emergency plans to satisfy Regulatory Guide 1.101, with special attention to the development of uniform action level criteria based on plant parameters.

- (2) Assure the implementation of the related recommendations of the NRR Lessons Learned Task Force involving instrumentation to follow the course of an accident and relate the information provided by this instrumentation to the emergency plan action levels. This will include instrumentation for post-accident sampling, high range radioactivity monitors, and improved in-plant radioiodine instrumentation. The implementation of the Lessons Learned recommendation on instrumentation for detection of inadequate core cooling will also be factored into the emergency plan action level criteria.
- (3) Determine that an Emergency Operations Center for Federal, State and local personnel has been established with suitable communications to the plant, and that upgrading of the facility in accordance with the Lessons Learned recommendation for an in-plant technical support center is underway.
- (4) Assure that improved licensee offsite monitoring capabilities (including additional TLD's or equivalent) have been provided for all sites.
- (5) Assess the relationship of State/local plans to the licensee's and Federal plans so as to assure the capability to take appropriate emergency actions. Assure that this capability will be extended to a distance of 10 miles as soon as practical, but not later than January 1, 1981. This item will be performed in conjunction with the Office of State Programs and the Office of Inspection and Enforcement.
- (6) Require test exercises of approved Emergency Plans (Federal, State, local, licensees), review plans for such exercises, and participate in a limited number of joint exercises. Tests of licensee plans will be required to be conducted as soon as practical for all facilities and before reactor startup for new licensees. Exercises of State plans will be performed

in conjunction with the concurrence reviews of the Office of State Programs. Joint test exercises involving Federal, State, local and licensees will be conducted at the rate of about 10 per year, which would result in all sites being exercised once each five years.

The staff review will be accomplished by about 6 review teams, similar to the concept used to assure suitable implementation of the physical security provisions of 10 CFR 73.55. As a minimum, the teams will consist of a team leader from NRR, a member from Los Alamos Scientific Lab (LASL) and, at least for field visits, a member from the IE Regional office. LASL will be used as the source of non-NRC team members because of the expertise gained and familiarity with the plants acquired during the physical security reviews. The Division of Operating Reactors will have the responsibility for completing these reviews for both operating reactors and near-term OL's. J. R. Miller, Assistant Director, DOR will be responsible for implementation of the program. General policy and technical direction will be provided by Brian Grimes, Assistant Director, DOR.

The first sites to be reviewed by the teams will be those scheduled for operating licenses within the next year and those sites in areas of relatively high population. Major milestones for the program are being developed and will include regional meetings with licensees to discuss the program, site visits by the review team, and meetings with local officials.

Coordination: This action plan has been discussed with the Task Force on Emergency Planning and the Task Force Chairman, T. F. Carter, has advised that the Task Force deliberations to date have indicated no reason why NRR should not proceed. The Office of State Programs concurs in this plan. The Office of Inspection and Enforcement concurs in the plan.

The Commissioners

- 4 -

NRR expects to perform this task without augmentation of resources beyond those authorized for FY79 and FY80.

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure:
Emergency Preparedness Improvements
for Operating Plants and Near
Term OL's

DISTRIBUTION

Commissioners
Commission Staff Offices
Exec Dir for Operations
ACRS
Secretariat

ENCLOSURE NO. 1

EMERGENCY PREPAREDNESS IMPROVEMENTS

AND COMMITMENTS REQUIRED FOR OPERATING PLANTS AND NEAR TERM OL'S

<u>Item</u>	<u>Implementation Category^{1/}</u>
1. Upgrade emergency plans to Regulatory Guide 1.101 with special attention to action level criteria based on plant parameters.	A ¹
2. Implement certain short term actions recommended by Lessons Learned task force and use these in action level criteria. ^{2/}	
2.1.8(a) Post-accident sampling	
Design review complete	A
Preparation of revised procedures	A
Implement plant modifications	B
Description of proposed modification	A
2.1.8(b) High range radioactivity monitors	B
2.1.8(c) Improved in-plant iodine instrumentation	A
3. Establish Emergency Operations Center for Federal, State and Local Officials	
(a) Designate location and alternate location and provide communications to plant	A ¹
(b) Upgrade Emergency Operations Center in conjunction with in-plant technical support center	B

^{1/}

Category A: Implementation prior to OL or by January 1, 1980 (see NUREG-0578).

Category A¹: Implementation prior to OL or by mid-1980.

Category B: Implementation by January 1, 1981.

^{2/}

The implementation of the Lessons Learned task force recommendation item 2.1.3(b), instrumentation for detection of inadequate core cooling, will also be factored into the action level criteria.

<u>Item</u>	<u>Implementation Category</u>
4. Improve offsite monitoring capability	A ¹
5. Assure adequacy of State/local plans	
(a) Against current criteria	A ¹
(b) Against upgraded criteria	B
6. Conduct test exercises (Federal, State, local, licensee)	
(a) Test of licensees emergency plan	A ¹
(b) Test of State emergency plans	A ¹
(c) Joint test exercise of emergency plans (Federal, State, local, licensee)	
New OL's	B
All operating plants	Within 5 years

RECOMMENDATION 1

INCREASE MINIMUM EXPERIENCE REQUIREMENTS FOR SENIOR OPERATOR APPLICANTS

**4 YEARS OF OPERATING EXPERIENCE
2 YEARS NUCLEAR — 6 MONTHS ON SITE**

- **NUCLEAR PLANT STAFF ENGINEER OR**
- **CONTROL ROOM OPERATOR**
- **2 YEARS MAY BE ACADEMIC**

RECOMMENDATION 2

***SENIOR OPERATOR APPLICANTS
MUST HOLD AN OPERATOR
LICENSE FOR SIX MONTHS***

RECOMMENDATION 3

***MORE SPECIFIC TRAINING
REQUIREMENTS FOR HOT
LICENSE APPLICANTS***

RECOMMENDATION 4

***REQUIRE SIMULATOR
TRAINING FOR HOT
LICENSE APPLICANTS***

RECOMMENDATION 5

***MORE FREQUENT AUDITING OF
TRAINING PROGRAMS,
INCLUDING ADMINISTRATION OF
SOME CERTIFICATION
EXAMINATIONS***

RECOMMENDATION 6

***REQUIRE INSTRUCTORS TO
HOLD SENIOR OPERATOR
LICENSES***

RECOMMENDATIONS 7, 8 AND 9

REQUALIFICATION PROGRAMS

- 1. REQUIRE ANNUAL SIMULATOR RETRAINING**
- 2. REQUIRE SPECIFIC EXERCISES**
- 3. NRC ADMINISTER SOME OF THE ANNUAL EVALUATIONS**

RECOMMENDATIONS 10, AND 13

NRC WRITTEN EXAMINATIONS

- A. INCREASE THE SCOPE TO INCLUDE
THERMODYNAMICS, HYDRAULICS AND
RELATED SUBJECTS**
- B. NEW PASSING GRADES FOR WRITTEN
EXAMINATIONS**
 - 80% OVERALL**
 - 70% EACH CATEGORY**
- C. INFORM FACILITY MANAGEMENT OF
RESULTS**

RECOMMENDATIONS 11, 12 AND 15

NRC OPERATING TESTS

- A. PART OF THE OPERATING TEST TO BE
ADMINISTERED ON A SIMULATOR**
- B. SENIOR OPERATORS TO BE
ADMINISTERED SIMULATOR OPERATING
TESTS**
- C. REVIEW ANSI/ANS 3.5-1979 "NUCLEAR
POWER PLANT SIMULATORS"**

APPENDIX C

VIEWGRAPHS

PRESENTATIONS TO ACRS SUBCOMMITTEE

ON IMPROVED SAFETY SYSTEMS

WASHINGTON, D. C.

JUNE 26, 1979

<u>APPENDIX C1</u>	-	<u>R. DISALVO OF NRC</u>
<u>APPENDIX C2</u>	-	<u>M. SILBERBERG OF NRC</u>
<u>APPENDIX C3</u>	-	<u>M. NORIN OF DOE AND</u> <u>D. DAHLGREN OF SANDIA</u>

DR. RAYMOND DISALVO
PROBABILISTIC ANALYSIS STAFF
OFFICE OF NUCLEAR
REGULATORY RESEARCH

CONCEPTS TO IMPROVE LWR SAFETY

PRESENTED TO THE IMPROVED SAFETY SYSTEMS
SUBCOMMITTEE OF THE ADVISORY COMMITTEE
ON REACTOR SAFEGUARDS

CONTENT OF PRESENTATION

ADMINISTRATIVE STATUS

NRC-DOE COORDINATION

TECHNICAL STATUS

PROGRAMS IN PLACE

PROGRAMS PENDING

PROGRAMS PLANNED

SPECIAL TOPICS

FY 1979 PROGRAMMING

COMMITTED

VENTED CONTAINMENT	0.3M
HUMAN ERROR SENSITIVITY STUDY	0.1M
SHUTDOWN HEAT REMOVAL	

PENDING

IN-PLANT ACCIDENT RESPONSE	0.2M
VALUE-IMPACT METHODS	0.1M
SHUTDOWN HEAT REMOVAL	0.1M

EFFECT OF TMI-2 ON PROGRAMMING

	1979		1980		1981	
	PRE	POST	PRE	REQUESTED	REQUESTED	AMENDED
ALTERNATE CONTAINMENT	0.3	0.3	0.3	0.6	0.6	0.8
ALTERNATE DECAY HEAT REMOVAL	0.2	0.1	0.2	0.4	0.4	0.4
ALTERNATE ECCS	--	--	--	0.3	0.3	1.0
HUMAN INTERACTION	--	0.3	0.4	2.1	2.3	2.7
ADVANCED SEISMIC DESIGN	--	--	--	0.3	0.4	1.0
SCOPING STUDIES	--	--	--	0.4	0.4	0.4
IMPROVED METHODOLOGY	0.3	0.1	0.1	0.3	0.3	0.3
	—	—	—	—	—	—
TOTAL (\$M)	0.8	0.8	1.0	4.4	4.7	6.6

HUMAN INTERACTION

TOPIC	FY 1979	FY 1980		FY 1981
		PRESIDENT'S	REVISED	REQUESTED
HUMAN ERROR SENSITIVITY	0.1	--	0.1	--
ACCIDENT MONITORING AND DIAGNOSTICS	0.1	0.2	0.8	0.9
REQUIREMENTS FOR IMPROVED INSTRUMENTATION	--	--	0.4	0.4
HUMAN INTERACTIONS REVIEW	0.1	0.1	0.1	0.1
CLASS 9 SIMULATOR CAPABILITY	--	--	0.3	0.4
SAFETY SYSTEM INTERLOCK	--	0.1	0.2	0.3
INFORMATION FLOW DURING REACTOR ACCIDENTS	--	--	0.2	0.2
TOTAL (\$M)	0.3	0.4	2.1	2.3

TABLE 3-1. CONCEPTS FOR THE IMPROVEMENT OF REACTOR SAFETY

RESEARCH TOPIC	SOURCE								QUALITATIVE ASSESSMENT MATRIX								
	ACRS NUREG-0392 ^a	NRC STAFF	CONSULTANTS	APS STUDY GROUP REPORT ^b	FORD FOUNDATION STUDY ^c	ECCS ACCEPTANCE CRITERIA ^{d,e}	ENVIRONMENTAL QUALITY LABORATORY ^f		RISK REDUCTION POTENTIAL			GENERIC APPLICABILITY			ESTIMATED COST OF IMPLEMENTATION		
									H	M	L	H	M	L	H	M	L
PLANT SURVEILLANCE AND OPERATION																	
1. NDE and On-Line Monitoring		O	O	O						O	→	O					O
2. Improved Plant Controls		O	O	O	O						O	←	O				O
3. Improved In-Plant Accident Response		O	O	O					O	→		O					O
4. Reduced Occupational Exposure	O	O	O							O	→	←	O				O
SAFETY SYSTEMS																	
5. Alternate Emergency Core Cooling Concepts	O	O	O	O	O	O	O			O			O			←	O
6. Alternate Decay Heat Removal Concepts	O	O	O	O					O			←	O				O
7. Alternate Containment Concepts	X	O	O	O	O				O			O	→	→	←	O	
8. Improved Reactor Shutdown Systems			O		O					O			O	→		O	
9. Reactor Vessel Rupture Control	R				O						O		O		←	O	
10. Core Retention Measures	O		O	O							O		O	→		O	→
11. Equipment for Reducing Radioactivity Releases	R	O	O								O	←	O			←	O
PLANT CONFIGURATION AND DESIGN																	
12. Advanced Seismic Designs	O	O	O						O	→			←	O	←	O	
13. Improved Plant Layout and Component Protection	O	O	O							O	→		←	O		←	O
14. Protection Against Sabotage	O	O	O	O	O								O			←	O
SITING AND EMERGENCY RESPONSE																	
15. New Siting Concepts	X	O	O	O						O			←	O		O	→
16. Improved Off-Site Emergency Response Planning	O		O	O	O					←	O	O					O

^a The symbol R indicates recommendations made in ACRS reports but not identified in NUREG-0392; X indicates that recommendation could be interpreted as being in more than one concept category.

^b Published in *Reviews of Modern Physics*, Vol. 47, Supplement No. 1, Summer 1975.

^c Published as *Nuclear Power-Issues and Choices*, Ballinger Publishing Company, Cambridge, Mass., 1977.

^d Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, U.S. Atomic Energy Commission, Washington, D.C., Docket No. RM-50-1, December 1973.

^e ECCS was the principal focus of this document.

^f F.C. Finlayson, Assessment of Emergency Core Cooling System Effectiveness for Light Water Nuclear Power Reactors, EOL Report No. 9, Environmental Quality Laboratory, California Institute of Technology, May 1975.

STATUS OF IMPROVED REACTOR SAFETY RESEARCH

<u>TOPIC</u>	<u>WORK SCOPE</u>	<u>PROPOSALS</u>	<u>CONTRACT</u>
CONTAINMENT			
VENTED	COMPLETE	SELECTED	SANDIA
OTHERS	COMPLETE	BEING SOLICITED	--
DECAY HEAT REMOVAL			
US	COMPLETE	SELECTED	SANDIA
FOREIGN	BEING DEVELOPED	BEING EVALUATED	----
ALTERNATE ECCS	DRAFT	--	--
HUMAN INTERACTION			
HUMAN ERROR SENSITIVITY	COMPLETE	SELECTED	BNL
MONITORING AND DIAGNOSTICS	DRAFT	SELECTED	PENDING
INSTRUMENTATION REQUIREMENTS	--	--	--
HUMAN INTERACTIONS REVIEW	COMPLETE	SELECTED	PENDING
SAFETY INTERLOCKS	DRAFT	--	--
CLASS 9 SIMULATOR	BEING DEVELOPED	--	--
INFORMATION FLOW	--	--	--

STATUS OF IMPROVED REACTOR SAFETY RESEARCH (CONT.)

<u>TOPIC</u>	<u>WORK SCOPE</u>	<u>PROPOSALS</u>	<u>CONTRACT</u>
SEISMIC DESIGN			
ENERGY ABSORBING DEVICES	DRAFT	BEING EVALUATED	--
TOPOGRAPHIC MODIFICATIONS	BEING DEVELOPED	--	--
ISOLATION	BEING DEVELOPED	--	--
SCOPING STUDIES	--	--	--
IMPROVED METHODOLOGY	COMPLETE	SELECTED	PENDING

INPUTS TO FUTURE PLANNING

ACRS RECOMMENDATIONS

LESSONS LEARNED TASK FORCE

PRESIDENT'S COMMISSION

STAFF, CONTRACTORS AND PUBLIC

NUCLEAR SAFETY ANALYSIS CENTER

OPERATING EXPERIENCE

CONFIRMATORY RESEARCH

SUMMARY OF ADMINISTRATIVE STATUS

FY 1979

- \$0.4M COMMITTED
- \$0.4M PENDING CONGRESSIONAL APPROVAL

FY 1980

- \$1.0M IN PRESIDENT'S BUDGET
- \$4.4M FLOOR IN PROPOSED AUTHORIZATION
- \$1.0M IN PROPOSED APPROPRIATION
- WORK SCOPES IN VARIOUS STAGES

FY 1981

- \$4.7M REQUESTED (ASSUMES \$1.0M IN 1980)
- \$6.6M AMENDED (ASSUMES \$4.4M IN 1980)

NRC-DOE COORDINATION ON IMPROVED SAFETY

STAFF CONTACTS

EXCHANGE DOCUMENTS

WORK PLANS

PROGRESS REPORTS

BUDGET PLANNING

REVIEW GROUPS

COORDINATING COMMITTEE

CHARTER OF NRC RESEARCH TO IMPROVE LWR SAFETY

SAFETY-MOTIVATED

DEVELOP AND EVALUATE CONCEPTS

ASSESS FEASIBILITY

NET EFFECT ON RISK

VALUE/IMPACT

PROPOSE NEW OR REVISED REQUIREMENTS

FUNCTIONAL

PERFORMANCE

SAFETY DESIGN

CONTENT OF PRESENTATION

ADMINISTRATIVE STATUS

NRC-DOE COORDINATION

TECHNICAL STATUS

PROGRAMS IN PLACE

PROGRAMS PENDING

PROGRAMS PLANNED

SPECIAL TOPICS

VENTED CONTAINMENT

CONTRACTOR: SANDIA LABORATORIES

OBJECTIVE: TO PERFORM ENGINEERING INVESTIGATIONS OF VENT-FILTERED CONTAINMENT CONCEPTS, RESULTING IN THE DEFINITION OF SYSTEM DESIGN REQUIREMENTS AND THE RISK REDUCTION VALUE AND COST IMPACT ASSOCIATED WITH THESE REQUIREMENTS

FUNDS: FY 1979 - 300K
FY 1980 - 300K

VENTED CONTAINMENT

<u>TASK</u>	<u>SCHEDULE</u>
DRAFT PROGRAM PLAN	COMPLETED
DEVELOP DESIGN CONCEPTS	2/80
PERFORM ENGINEERING ANALYSES	6/80
INTERIM REPORT	6/80
DEVELOP ANALYTICAL MODELS, CALCULATE CONSEQUENCES	3/81
PERFORM VALUE-IMPACT ASSESSMENT	4/81
PROPOSE DESIGN REQUIREMENTS	5/81
FINAL REPORT	6/81

ALTERNATE SHUTDOWN HEAT REMOVAL CONCEPTS

CONTRACTOR: SANDIA LABORATORIES

OBJECTIVE: TO DEVELOP DESIGN REQUIREMENTS WHICH ENHANCE
THE RELIABILITY AND AVAILABILITY OF DECAY HEAT
REMOVAL SYSTEMS AND TO ASSESS THE VALUES AND
IMPACTS OF IMPLEMENTING THESE REQUIREMENTS

FUNDS: FY 1979 - 100K
FY 1980 - 300K
FY 1981 - 200K

ALTERNATE SHUTDOWN HEAT REMOVAL CONCEPTS

<u>TASK</u>	<u>SCHEDULE</u>
IDENTIFY CURRENT DESIGNS AND CRITERIA	9/79
IDENTIFY EVENTS REQUIRING OR THREATENING SHR OPERATION	2/80
DEVELOP SHR LOGIC MODELS	3/80
SELECT CONCEPTUAL DESIGN OPTIONS (DESIGN REQUIREMENTS)	9/80
SELECT VALUE AND IMPACT MEASURES	9/80
ANALYZE OPTIONS	4/81
DOCUMENT RESULTS	7/81

HUMAN ERROR SENSITIVITY ANALYSIS

CONTRACTOR: BROOKHAVEN

OBJECTIVE: TO IDENTIFY HUMAN ERRORS WHOSE REDUCTION.
WOULD MOST EFFECTIVELY REDUCE RISK

FUNDS: FY 1979 - 100K
FY 1980 - 100K

HUMAN ERROR SENSITIVITY ANALYSIS

<u>TASK</u>	<u>SCHEDULE</u>
CATEGORIZE HUMAN ERRORS IN WASH-1400	7/79
RANK THE CONTRIBUTIONS OF THESE ERRORS	10/79
VARY HUMAN ERROR RATES	4/80
RANK RISK REDUCTION POTENTIALS	9/80

PRESSURIZED WATER REACTOR's

HUMAN ERROR LIST FROM WASH-1400

DRAFT

Brookhaven National Laboratory
Associated Universities, Inc.
Upton, New York 11973

A. Azarinn
J.M. Dickey
A. Swoboda

DESCRIPTION	CAUSE	WHEN	WHERE	WHY	DETECTION	SYSTEM	COMPONENT	CODE	TABLE	FIG. (sheet)	UNAVAILABILITY/E
Operator error SIB1-B Block closed when SICS Act	C	POST	Control Rm.	0		SICS	No Component	MOUB038X	11 5-25	11 5-48	3.0x10 ⁻⁴ /3
Operator error pump turned off inadvertently	C	POST	Control Rm.	0		CSRS	Pump	DPH2A03X(B)	11 5-27	11 5-54	1.0x10 ⁻³ /3
Operator error fails to verify if pump is running	0	POST	Control Rm.	0		CSRS	No Component	DOUCS1AX(B)	11 5-27	11 5-55	1.0x10 ⁻² /3
Operator error inadvertently stops pump	C	POST	Control Rm.	0		CSRS	No Component	DOUCS3AX(B)	11 5-27	11 5-55	1.0x10 ⁻³ /3
Operator error fails to verify if pump is running	0	POST	Control Rm.	0		CSRS	No Component	DOUCS2AX(B)	11 5-27	11 5-55	1.0x10 ⁻² /3
Air vent failure inhibits flow thru heat exchange						CHRS	Manual Valve	XXV1A20X (B,C,D)	11 5-29	11 5-60	3x10 ⁻⁵ /3
Air vent failure inhibits flow thru heat exchange						CHRS	Manual Valve	KXV1A21X (B,C,D)	11 5-29	11 5-60	3x10 ⁻⁵ /3
MOV-SW-104A inadvertently closed	C					CHRS	Control Switch	KCS4A43X (B,C,D)	11 5-29	11 5-60	1x10 ⁻³ /3
Operator error - all heat exchanger air vents left closed	0					CHRS	Heat Exchanger	KHEVENTY	11 5-29	11 5-60	1x10 ⁻⁵ /3
NO 480V power due to breaker open						CHRS	Circuit Breaker	KCBHAPVY	11 5-30	11 5-61	E
MOV-SW-105A inadvertently closed	0					CHRS	Control Switch	KCS5A43X (B,C,D)		11 5-60	not listed
MOV in suction line not opened for recirculation because CLCS not reset	0		Control Rm.	0	Light	LPRS	Motor Oper. Valve	EMV0001X	11 5-32	11 5-64(1)	1.0x10 ⁻⁵ /3
Operator error MOVA05(B05) not opened for recirculation	0	POST	Control Rm.	0	Light	LPRS	Motor Oper. Valve	EMVA005X(B)	11 5-32	11 5-64(1&2)	E
Operator error inadvertently loses MOV 1890C	C	POST	Control Rm.	0	Light	LPRS	Motor Oper. Valve	EMV890CX	11 5-32	11 5-64(3)	1.0x10 ⁻³ /3

REDUCING OPERATOR CONTRIBUTION TO RISK

WHAT IS THE STATUS OF THE PLANT?

HOW IS STATUS BEST DISPLAYED?

WHAT DOES THE DISPLAY MEAN TO THE OPERATOR?

- BASIC KNOWLEDGE
- TRAINING
- PREVIOUS EXPERIENCE
- PHYSICAL AND MENTAL AWARENESS

WHAT SHOULD THE OPERATOR DO?

- DIAGNOSTIC AIDS
- PROCEDURES
- CORRECTIVE ACTION AIDS

Experimental Control Room Facility

2 Physical Lay-Out Format - Leakage Location

③ Circuit Diagram - Survey of Halden
Reactor Main Circuits C1-24

IN-PLANT ACCIDENT RESPONSE

CONTRACTOR: ORNL (PENDING)

OBJECTIVE: EVALUATE THE FEASIBILITY OF AND PROPOSE
REQUIREMENTS FOR IMPROVED SYSTEMS TO
ASSIST REACTOR OPERATORS

FUNDS: FY 1979 - 200K
FY 1980 - 200K

IN-PLANT ACCIDENT RESPONSE

<u>TASK</u>	<u>SCHEDULE (MONTHS)</u>
REVIEW INFORMATION AVAILABLE TO OPERATOR	+4
REVIEW SYSTEMS FOR SAFETY INTERLOCK POTENTIAL	+4
IDENTIFY STATE-OF-ART IN COMPUTERIZED MONITORING AND DIAGNOSTICS	+6
PROPOSE PRELIMINARY GENERAL DESIGN REQUIREMENTS	+8

VALUE-IMPACT METHODOLOGY

CONTRACTOR: BATTELLE NORTHWEST LABORATORY (PENDING)

OBJECTIVE: TO DEVELOP AND APPLY METHODS FOR ASSESSING
THE VALUES AND IMPACTS OF PROPOSED IMPROVED
SAFETY CONCEPTS

FUNDS: FY 1979 - 100K
FY 1980 - 300K

VALUES AND IMPACTS

WHO BENEFITS?

**PUBLIC
PLANT PERSONNEL
NRC
LICENSEE
OTHERS**

HOW?

REDUCED RISK
• ACUTE
• CHRONIC
• PROPERTY
REDUCED LICENSING TIME
REDUCED COST

WHO PAYS?

**PUBLIC
PLANT PERSONNEL
NRC
LICENSEE
OTHERS**

FOR WHAT?

INCREASED RISK
INCREASED LICENSING TIME
INCREASED COST
• R&D
• LICENSING
• CONSTRUCTION
• OPERATION
• MAINTENANCE

VALUE-IMPACT METHODOLOGY

OBJECTIVE

**TO DEVELOP AND APPLY METHODS FOR ASSESSING THE VALUES
AND IMPACTS OF PROPOSED CONCEPTS FOR IMPROVING REACTOR
SAFETY**

SCOPE

DEVELOP METHODOLOGY

**DEFINE CATEGORIES AND MEASURES OF VALUE & IMPACT
DEFINE SPECTRUM OF ACCIDENTS TO BE CONSIDERED
DEFINE BASELINE CONDITIONS
DEVELOP SYSTEM TO ASSESS RISK REDUCTION POTENTIAL
AND OTHER KEY VALUES AND IMPACTS
DEVELOP TECHNIQUES TO TEST SENSITIVITY
DEVELOP FORMATS FOR PROVIDING INPUT AND PRESENTING
RESULTS
CODIFY METHOD OF CHOICE**

APPLY METHODOLOGY

**SPECIFY DATA NEEDS FOR ASSESSMENTS OF INDIVIDUAL CON-
CEPTS
ASSIST OTHERS IN ASSESSMENTS OF INDIVIDUAL CONCEPTS
PERFORM COMPARATIVE ASSESSMENTS OF CONCEPTS**

ALTERNATE CONTAINMENT CONCEPTS

OBJECTIVE: ASSESS THE VALUES AND IMPACTS OF ALTERNATE
CONTAINMENT CONCEPTS

SCOPE:

- o REVIEW PREVIOUS ANALYSES, CONCEPTUAL DESIGNS
AND PERTINENT EXPERIMENTAL INFORMATION
- o ASSESS FEASIBILITY
- o ASSESS RELATIVE VALUES AND IMPACTS

EXAMPLES OF ALTERNATE CONTAINMENT CONCEPTS

- o VENTED FILTERED CONTAINMENT
- o LARGER VOLUMES
- o HIGHER PRESSURE CAPABILITIES
- o REPRESSURIZATION
- o PASSIVE CONTAINMENT SYSTEMS
- o HYDROGEN RECOMBINERS
- o REDUCED INITIAL OPERATING PRESSURES
- o AUXILIARY BUILDING SHIELDING AND FILTERS
- o MITIGATION OF STEAM EXPLOSION
- o MITIGATION OF HYDROGEN EXPLOSION OR BURNING
- o MOLTEN CORE RETENTION DEVICES
- o VARIATIONS IN BASE MAT DESIGN
- o IMPROVEMENTS IN PENETRATION DESIGN
- o IMPROVEMENTS IN FISSION PRODUCT CONTROL

ALTERNATE EMERGENCY CORE COOLING CONCEPTS

OBJECTIVE

**ASSESS THE VALUES AND IMPACTS OF ALTERNATE ECCS HAVING
EASILY ANALYZABLE AND CLEARLY DEMONSTRABLE CAPABILITY
FOR CORE COOLING**

SCOPE

IDENTIFY CONCEPTS

PERFORM PRELIMINARY VALUE/IMPACT

REVIEW EXISTING AND PLANNED ANALYSES AND EXPERIMENTS

RELAP AND ADVANCED CODES

SEMISCALE

LOFT

2D/3D

TLTA

**ANALYZE PERFORMANCE OF ALTERNATE ECCS WITH ADVANCED
CODES**

**PROPOSE ADDITIONAL EXPERIMENTS AS NECESSARY TO VERIFY
MOST PROMISING CONCEPT(S)**

PERFORM DETAILED VALUE/IMPACT

ALTERNATE EMERGENCY CORE COOLING INJECTION POINTS SUGGEST IMPROVED COOLING

ADVANCED SEISMIC DESIGNS

OBJECTIVE

**ASSESS THE VALUES AND IMPACTS OF DESIGNS TO REDUCE
THE CONTRIBUTION TO RISK FROM SEISMIC EVENTS**

SCOPE

**REVIEW CANDIDATE CONCEPTS TO DETERMINE FEASIBILITY
STRENGTHEN CURRENT DESIGNS
INCREASED ENERGY ABSORPTION CAPABILITY
COMPONENT ISOLATION
FOUNDATION ISOLATION**

DEFINE PRELIMINARY SYSTEM DESIGN REQUIREMENTS

IMPROVE ANALYTICAL MODELS

PERFORM PRELIMINARY VALUE – IMPACT ASSESSMENT

CONDUCT VERIFICATION EXPERIMENTS AS NEEDED

CONCEPTS FOR ATTENUATION OF
SEISMIC EFFECTS FOR
NUCLEAR POWER PLANT CONTAINMENTS

by

Philip J. Richter
Robert P. Kennedy

As part of a Sandia Laboratories effort on means of increasing safety of nuclear power plant containment structures and equipment, this study reviews enhancement of seismic safety by methods of response attenuation. After a quick look at several concepts, the effort provides some preliminary development of four selected concepts. The attenuation method selected as most promising for both the containment and equipment is horizontal isolation utilizing low friction bearings plus hysteretic energy absorbing devices. This method is promising because of its simplicity, but much development to assure proper design and function is required prior to deployment.

prepared for
SANDIA LABORATORIES
Albuquerque, New Mexico

July 1977

EDAC

ENGINEERING DECISION ANALYSIS COMPANY, INC

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2400 MICHELSON DRIVE

BURNITZSTRASSE 34

PALO ALTO CALIF 94306

IRVINE, CALIF 92715

6 FRANKFURT 70 W GERMANY

FIGURE 5-1. CONCEPT A-1 BASE MOUNTED SEISMIC ISOLATION

TABLE 5-1
CONCEPT A-1 BASE MOUNTED SEISMIC ISOLATION
ADVANTAGES AND DISADVANTAGES

ADVANTAGES -

- . Major Components - Slide Bearings, Isolators, and Dampers have Considerable Development and Some Analogous Deployment
- . Relatively Simple Overall Design Concept
- . Can Reduce Seismic Forces to Low Levels for Entire Structure and Internal Equipment
- . May Minimize Additional Relative Displacements
- . High Degree of Redundancy and Overall Structural Safety
- . Applicable to Entire Nuclear Plant as well as Containment Alone
- . Potential for Cost Effectiveness after Development

DISADVANTAGES

- . Because of Massiveness, Weight and Size, Overall Concept Requires Extensive Development, Design and Verification
- . Lateral Isolators such as TOR-SHOKS with Bilinear Characteristics and Large Damping, Require Development for Higher Force Levels
- . Possible Large Relative Displacement May Require Considerable Modification to Piping Design for Flexibility and Possibly Rearrangement of Overall Architectural Layout to Provide Additional Space
- . Innovative Concept Requires Considerable Time and Verification to Assure Function and to Gain Acceptance by Regulatory Commission

FIGURE 5-12. CONCEPT B-2 ENERGY ABSORPTION - STEAM GENERATOR

TABLE 5-4
ADVANTAGES AND DISADVANTAGES
CONCEPT B-2 ENERGY ABSORPTION
STEAM GENERATOR

ADVANTAGES

- . Simplicity of Concept
- . Can be Retrofitted in Some Cases
- . May Minimize Relative Displacement
- . Relatively Low Cost after Development
- . Concept in General Well Proven

DISADVANTAGES

- . Only Efficient for Fairly Flexible Equipment
- . Requires Relatively Large Surface Area of Equipment for Mounting
- . May Require Adjustment of Surrounding Layout to Accommodate Mounting Space
- . Requires Additional Verification for Seismic Environment

SCOPING STUDIES OF OTHER CONCEPTS

OBJECTIVE

**TO IDENTIFY THE NEED FOR FURTHER RESEARCH AMONG
ALTERNATIVE CONCEPTS SUGGESTED TO IMPROVE SAFETY**

SCOPE

REVIEW AND EVALUATE ADDITIONAL CONCEPTS

**OFF-SITE EMERGENCY RESPONSE*
PROTECTION AGAINST SABOTAGE*
NDE AND ON-LINE MONITORING*
NEW SITING CONCEPTS*
REDUCED OCCUPATIONAL EXPOSURE*
IMPROVED REACTOR SHUTDOWN SYSTEMS*
IMPROVED PLANT LAYOUT
IMPROVED PLANT CONTROL
CORE RETENTION MEASURES*
REDUCED RADIOACTIVITY RELEASES
REACTOR VESSEL RUPTURE CONTROL**

***WORK IN THESE AREAS IS PART OF NRC'S ONGOING PROGRAMS**

SUMMARY OF TECHNICAL STATUS

- o PRIORITIES REVISED TO REFLECT TMI-2 CONCERNS
- o WORK INITIATED ON HIGHEST PRIORITY TOPICS
 - VENTED CONTAINMENT
 - HUMAN INTERACTION
 - ALTERNATE DECAY HEAT REMOVAL
- o RATE, DEPTH, AND BREADTH OF FUTURE PROGRESS DEPENDS ON AVAILABILITY OF FUNDS

NRC CORE RETENTION RESEARCH

PRESENTATION BY
M. SILBERBERG, RSR
TO ACRS SUBCOMMITTEE
ON IMPROVED REACTOR SAFETY RESEARCH
JUNE 26, 1979

NRC CORE RETENTION RESEARCH

- 1978 ACRS REVIEW / EVALUATION OF RES PROGRAM

"ACRS RECOMMENDS FURTHER THAT EMPHASIS BE GIVEN TO . . .
SCOPING STUDIES ON THE TOPICS RELATING TO PREVENTION OR
MITIGATION OF THE OFFSITE CONSEQUENCES RESULTING FROM
POSTULATED CORE MELT ACCIDENTS VIA LIQUID PATHWAYS"

CURRENT NRC CORE MELT RESEARCH (SANDIA)

LWR (WRSR)

- OBJECTIVE - IMPROVED MODELS FOR RISK ANALYSIS
- SCOPE
 - MELT / CONCRETE INTERACTIONS
 - INTERACTION MODELING (INTER / CORCON)
 - MODEL VERIFICATION (EXCHANGE WITH FRG - KFK)

CURRENT NRC CORE MELT RESEARCH (SANDIA)

ADVANCED REACTOR (ARSR)

- OBJECTIVE

- DATA BASE FOR CONTAINMENT CODE DEVELOP. / VERIFICATION
- CORE RETENTION SYSTEM ASSESSMENT
- NRR USER NEEDS (CRBR - 1976; FFTF - 1978)

- SCOPE

- MELT / CONCRETE INTERACTIONS (SMALL AND LARGE SCALE)
- MELT / RETENTION MATERIAL INTERACTIONS
- LARGE FIELD SCALE FACILITY DEVELOPMENT (100 - 500 KG)
- ADVANCED INSTRUMENTATION - QUANTIFICATION

NRC / ARSR CORE RETENTION RESEARCH

- OBJECTIVE

- SCOPING STUDIES TO IDENTIFY IMPORTANT PHENOMENA FOR RETENTION MATERIALS
- QUANTITATIVE DATA BASE FOR CANDIDATE RETENTION MATERIALS EVALUATION
- ESTABLISH FRAMEWORK FOR INTERACTION MODELS

- SCOPE

- LARGE SCALE (200 KG) SCOPING TESTS WITH MOLTEN S.S. (1,700°C)
- SMALL SCALE TESTS WITH STEEL & UO_2 / STEEL MELTS (PARAMETERS)
- SUPPORTING SEPARATE EFFECTS TESTS (CHEMICAL ATTACK)

CANDIDATE CORE RETENTION MATERIALS

- CRUCIBLE MATERIALS

- MgO^*

- UO_2

- ZrO_2

- TAC

- SACRIFICIAL MATERIALS

- BORAX*

- LEAD

- Fe_3O_4

- MISC. MATERIALS

- FIREBRICK*

- HIGH ALUMINA CEMENT*

KEY QUESTIONS FOR CORE RETENTION MATERIALS (MgO)

- MECHANISM AND RATE OF MELT ATTACK
- MELT PENETRATION OF JOINTS
- EXFOLIATION OF LAYERS OF BRICK
- SLAG-LINE ATTACK
- HEAT FLUX DISTRIBUTION
- CREEP AND THERMAL SHOCK
- WATER FLOODING

FUTURE NRC CORE RETENTION RESEARCH

- GENERIC LARGE SCALE SUSTAINED MELT TESTS (500KG)
 - QUANTITATIVE
 - ENGINEERING FEATURES
- SUPPORT FOR NRR REVIEW OF FNP
- ASSESS RISK REDUCTION POTENTIAL

RELATED NRR USER REQUESTS

- RR-NRR-76-2 (MARCH 1976)
 - CRBR CORE DEBRIS RETENTION CAPABILITY
 - MgO
- RR-NRR-76-2 MODIFICATION (MARCH 1978)
 - FFTF CONTAINMENT MARGIN VERIFICATION
 - FIREBRICK
 - GENERIC DIRECTION
- RR-NRR-79-10 (APRIL 1979)
 - LICENSING REVIEW OF FLOATING NUCLEAR PLANT (MgO)
 - GENERIC INTEREST

NRC STAFF REQUIREMENTS FOR FNP RETENTION DEVICES

- SHALL PROVIDE INCREASED RESISTANCE TO CORE DEBRIS MELT-THROUGH
- SHALL NOT REACT WITH CORE DEBRIS TO FORM LARGE VOLUMES OF GAS
- SHALL BE AT LEAST AS THICK AS CURRENT CONCRETE PAD (4 FT.)
AND AS THICK AS PRACTICABLE
- SHALL NOT COMPROMISE SAFETY

NRR USER REQUEST
FLOATING NUCLEAR PLANT

- CONFIRM FEASIBILITY OF REFRACTORY MATERIAL RETENTION DEVICE
- KEY TEST NEEDS
 - QUASI-STEADY STATE CONDITIONS (SUSTAINED HEATING)
 - SCALING (GEOMETRY, SIZE)
 - EXAMINE MgO + ONE OTHER ATTRACTIVE CANDIDATE
 - ~ 3 YR. TIMEFRAME FOR MgO
- RES DEVELOPING RESPONSE

FLOATING NUCLEAR PLANT

C2-12

DOE LWR SAFETY TECHNOLOGY PROGRAM

PRESENTATION TO ACRS, JUNE 26, 1979

INTRODUCTORY REMARKS

M. P. NORIN, DOE

PROGRAM DESCRIPTION

D. A. DAHLGREN, SANDIA LABORATORIES

DOE LWR SAFETY TECHNOLOGY PROGRAM

CHARTER

ALL SAFETY RELATED CONCERNS

DOE LWR SAFETY TECHNOLOGY PROGRAM

CURRENT PLAN

IMPROVED SYSTEMS

MAN-MACHINE INTERACTION

RISK METHODS UTILIZATION

SAFETY DATA

EXPANDED PROGRAM

- o TMI EMPHASIS

- o ADD

- UTILITY TRAINING

- EMERGENCY AND RECOVERY PROCEDURES

- TMI EXAMINATION AND ANALYSIS

DOE LWR SAFETY PROGRAM

ACTIVITIES 3/79 TO 6/79

CONTINUE IMPLEMENTING FY79 PROGRAM

COMPLETE DRAFT PROGRAM PLAN

STUDY THREE MILE ISLAND

REVISE PROGRAM

DOE LWR SAFETY TECHNOLOGY PROGRAM

CURRENT PROGRAM

		STATUS	
		<u>3/79</u>	<u>6/79</u>
RISK METHODS UTILIZATION			
	R&D SELECTION METHODOLOGY	FUNDED	FUNDED
	QUANTITATIVE METHODS FOR DESIGN DECISIONS	STUDY	RFP OUT
	DATA BASE DESIGN	FUNDED	FUNDED
	ACCEPTABILITY CRITERIA	FUNDED	FUNDED
	RELIABILITY & SAFETY METHODS DEVELOPMENT	FUNDED	FUNDED
	SAFETY/RELIABILITY/DESIGN INTEGRATION	FUNDED	FUNDED
TECHNOLOGY DEVELOPMENT			
	ACCIDENT INITIATOR FORMATION	RFP OUT	FUNDED
	VALVE IMPROVEMENTS	WRITE RFP	FUNDED
	INELASTIC BEHAVIOR	WRITE RFP	FUNDED
	FIRE SAFETY	FUNDED	FUNDED
	IMPROVED MAINTENANCE DESIGN	RFP OUT	FUNDED
	IMPROVED CONTAINMENT	FUNDED	FUNDED
MAN-MACHINE INTERACTIONS			
	ADVANCED MONITORING & CONTROL		
	PROGRAM DEVELOPMENT	FUNDED	FUNDED
	EXPLORATORY CONTROL	FUNDED	FUNDED
	IMPROVED MAINTENANCE		
SAFETY DATA			
	FUEL DAMAGE LIMITS/DNBR LIMIT	STUDY	FUNDED
	LONG LIFE FUEL DESIGN SAFETY	STUDY	HOLD
	UNRESOLVED SAFETY ISSUES	STUDY	FUNDED

C3-6

LWR Safety Technology Program Structure

RISK METHODS

SYSTEMS AND COMPONENT FAILURE DATA COLLECTION AND
DISSEMINATION

SYSTEMS ANALYSIS INCLUDING PARTIAL AND INTERMITTENT
SYSTEMS OPERATIONS

FACTORING HUMAN ERROR INTO ANALYSES

REVIEWING ACCIDENT ANALYSES TO IDENTIFY GENERAL
CLASSES OF ACCIDENTS WHICH HAVE NOT HERETOFORE
BEEN CONSIDERED IN THE LICENSING PROCESS

IMPROVED SAFETY SYSTEMS

SIMPLIFIED PLANT LAYOUTS TO REDUCE THE PLANT'S SENSITIVITY TO COMMON-CAUSE ACCIDENT INITIATORS, SUCH AS INCORRECT MAINTENANCE ACTIVITIES, ADVERSE ENVIRONMENTAL CONDITIONS, AND FIRES: AND

UNIQUE CONTAINMENT SYSTEMS

IMPROVED SAFETY VALVES

IMPROVED SHUTDOWN HEAT REMOVAL SYSTEMS

PUMP APPLICATION AND PERFORMANCE UNDER EMERGENCY CONDITIONS

CONTAINMENT ISOLATION RESPONSE

HYDROGEN RECOMBINER REQUIREMENTS

IMPROVED COMPONENT RELIABILITY UNDER A BROAD SPECTRUM OF ENVIRONMENTS AND OPERATING NEEDS

SYSTEMS INTERACTIONS, PARTIAL AND INTERMITTENT OPERATION

SYSTEMS/COMPONENT/EQUIPMENT QUALIFICATION AND OPERATION IN ACCIDENT ENVIRONMENTS

EQUIPMENT QUALIFICATION FOR LONG TERM RADIATION

MAN-MACHINE INTERFACE

IMPROVED QA OF DESIGN AND OPERATING PROCEDURES

DIAGNOSTICS ASSISTANCE

SAFETY SYSTEMS STATUS BOARD

NORMAL, ACCIDENT AND POST ACCIDENT INSTRUMENTATION
REQUIREMENTS AND QUALIFICATIONS

HUMAN ERROR AVOIDANCE

INTERLOCKS TO ASSURE NON-VIOLATION OF TECH SPECS

REMOTE OPERATION

SAFETY DATA

THREE MILE ISLAND FUEL/CORE AND HARDWARE EXAMINATION.
THIS ACITVITY WILL BE PURSUED ON A PRIORITY BASIS TO
ASSURE THAT VALUABLE DATA IS NOT LOST

PRIMARY SYSTEMS BEHAVIOR, NATURAL CIRCULATION WITH
AND WITHOUT FLOW BLOCKAGE, VOID GENERATION, NATURAL
CONVECTION LIMITS, ETC.

FISSION PRODUCT RELEASE AND METEOROLOGICAL MODEL
VALIDITY VERIFICATION

HYDROGEN EXPLOSIONS GENERATION AND PHYSICAL AND
CHEMICAL BEHAVIOR

UTILITY TRAINING PROGRAM

ACCIDENT RESPONSE, INCLUDING SIMULATOR TRAINING

INCREASED CONSCIOUSNESS OF SAFETY IMPLICATIONS
AT ALL ORGANIZATIONAL LEVELS

IMPROVED MAINTENANCE AND TEST PROCEDURES

PARTIAL SYSTEMS FAILURE AND USE OF ALTERNATIVE
SYSTEMS TO ACCOMPLISH OBJECTIVES

UPGRADED OPERATOR CERTIFICATION AND TRAINING

EMERGENCY AND RECOVERY MEASURES

DEVELOP DESIGN GUIDANCE TO FACILITATE DECONTAMINATION
AND ACCIDENT RECOVERY

DEVELOP PROCEDURES FOR INPLANT AND NATIONAL EMERGENCY
RESPONSE ORGANIZATIONS

DOE COORDINATION WITH NRC

DOCUMENTS

WORK STATEMENTS

PROGRAM MONTHLY REPORTS

MEETINGS

APPROXIMATELY 30 DURING PAST YEAR

APPENDIX D

VIEWGRAPHS

PRESENTATION OF R. MATTSON TO

NRC COMMISSIONERS ON TMI-2

LESSONS LEARNED TASK FORCE

WASHINGTON, D. C.

JUNE 25, 1979

TMI-2 LESSONS LEARNED TASK FORCE

- BEGAN WORK IN LATE MAY
- 22 PEOPLE - WIDE TECHNICAL EXPERTISE
- GENERAL SCOPE IS REACTOR LICENSING
- COORDINATING WITH OTHERS

LESSONS LEARNED APPROACH

- STARTED WITH MANY ISSUES
- CATEGORIZING IMPORTANT LESSONS
- INTEND TF COMPLETION SEPTEMBER 1

BASIC PREMISES

- CURRENTLY UNDERSTAND 4 CONTRIBUTORS

DESIGN

EQUIPMENT MALFUNCTION

HUMAN ERRORS

REGULATORY

SCOPE

- LESSONS LEARNED TF WILL SPEAK TO:
 - REACTOR OPERATIONS
 - LICENSEE TECHNICAL QUALIFICATIONS
 - ONSITE EMERGENCY PREPAREDNESS
 - DESIGN CRITERIA FOR SAFETY AND PROCESS EQUIPMENT
 - TRANSIENT AND ACCIDENT ANALYSIS
 - EVALUATION OF OPERATING EXPERIENCE
 - NRR EMERGENCY PREPAREDNESS

INTERIM CONCLUSIONS

- WORK OF B&O TASK FORCE JUDGED ADEQUATE TO ASSURE SAFETY OF PRESENT OPERATIONS, WITH SOME PROMPT ADDITIONS FROM LESSONS LEARNED
- OTHER SHORT TERM LESSONS LEARNED SHOULD BE IMPLEMENTED AS SOON AS PRACTICAL

INTERIM CONCLUSIONS; CONT'D

- SHORT TERM LESSONS PLUS BULLETINS
SUFFICIENT FOR NEAR TERM CP&OL.
- CONTINUING EVALUATION WILL YIELD MORE
ITEMS FOR EARLY IMPLEMENTATION AND
SOME FUNDAMENTAL ISSUES FOR LONG TERM
STUDY AND RULEMAKING

SHORT-TERM RECOMMENDATIONS--OPERATIONS

- REACTOR OPERATIONS MANAGEMENT
 - DEFINE AND IMPLEMENT COMMAND AND CONTROL FUNCTION
 - ADD A SHIFT SAFETY ENGINEER
 - DEFINE AND IMPLEMENT SHIFT TURNOVER PROCEDURE
- IN-PLANT EMERGENCY PREPARATIONS
 - LIMIT CONTROL ROOM ACCESS--OPERATIONS COMMAND
 - ESTABLISH ONSITE EMERGENCY OPERATIONS CENTER
 - ASSEMBLE AS-BUILT DRAWINGS AND RECORDS
- OPERATIONAL RELIABILITY AND QUALITY ASSURANCE
 - ESTABLISH CLEAR CRITERION FOR MINIMUM QUALITY
 - REQUIRE SHUTDOWN IF VIOLATED
 - REQUIRE THOROUGH INQUIRY BEFORE RESTART

SHORT-TERM RECOMMENDATIONS--DESIGN AND ANALYSIS

- PROVIDE EMERGENCY POWER FOR "PROCESS" EQUIPMENT
 - PRESSURIZER HEATERS
 - PORV AND BLOCK VALVES
 - PRESSURIZER LEVEL INDICATOR
- PERFORMANCE TESTING OF RELIEF AND SAFETY VALVES
- INFORMATION TO AID OPERATOR IN ACCIDENT
 - DIRECT INDICATION OF RELIEF AND SAFETY VALVE POSITION
 - INSTRUMENTATION FOR DETECTION OF INADEQUATE CORE COOLING

SHORT-TERM DESIGN AND ANALYSIS, CONT.

- CONTAINMENT ISOLATION

BACKFIT DIVERSE ACTUATION REQUIRED BY SRP 6.2.4

RETHINK ESSENTIAL AND NON-ESSENTIAL SYSTEMS

- POST-ACCIDENT CONTAINMENT HYDROGEN CONTROL

PROVIDE RELIABLE AND DEDICATED PENETRATIONS FOR PURGE AND RECOMBINER SYSTEMS

INERT ALL MARK I AND II CONTAINMENTS

PROVIDE CAPABILITY TO USE RECOMBINERS AT ALL PLANTS (MINORITY VIEW)

- RADIATION CONTROL FOR SYSTEMS OUTSIDE CONTAINMENT

PERFORM LEAK TEST AND PROVIDE TECH SPECS FOR SAFETY AND PROCESS SYSTEMS

PERFORM SHIELDING REVIEW FOR SAFETY AND PROCESS SYSTEMS

SHORT-TERM DESIGN AND ANALYSIS, CONT.

- AUXILIARY FEEDWATER SYSTEM RELIABILITY
 - REQUIRE AUTOMATIC ACTUATION
 - REQUIRE DIRECT FLOW INDICATION
- INSTRUMENTS TO FOLLOW ACCIDENT
 - IMPROVE POST-ACCIDENT SAMPLING
 - INCREASE RANGE OF EFFLUENT MONITORS
 - PROVIDE HI-RANGE CONTAINMENT MONITOR
 - IMPROVE IN-PLANT AND EFFLUENT IODINE MEASUREMENT
- ANALYSIS OF DESIGN AND OFF-DESIGN EVENTS FOR IMPROVED TRAINING AND PROCEDURES
 - SMALL BREAK LOCA
 - CORE UNCOVERING
 - MULTIPLE FAILURE EVENTS

FURTHER LESSONS TO BE LEARNED

- RELIABILITY GOALS

- SYSTEM AND COMPONENT RELIABILITY

- FREQUENCY OF CHALLENGE TO SAFETY SYSTEMS

- DEGRADED COOLING

- PREVENTION - DESIGN AND TRAINING

- MITIGATION - DESIGN AND TRAINING

- CORE COOLING SYSTEMS

- CONTAINMENT SYSTEMS

- RADWASTE AND EFFLUENT SYSTEMS

- SAFETY AND PROCESS SYSTEM DESIGN CLASSIFICATIONS

- RELIABILITY

- ENVIRONMENTAL QUALIFICATION

- POWER SUPPLY

FURTHER LESSONS TO BE LEARNED, CONT.

- OPERATIONAL SAFETY

- OPERATOR TRAINING AND LICENSING

- OPERATIONS TRAINING

- TECHNICAL QUALIFICATIONS

- ROLE OF SIMULATORS

- CONTROL ROOM DISPLAYS

- COMPUTER AIDED FAULT DIAGNOSIS

- HUMAN ENGINEERING

- EQUIPMENT STATUS MONITORING

- EMERGENCY PREPAREDNESS

- NRR TECHNICAL AND MANAGEMENT ROLES

- CRISIS MANAGEMENT PREPARATIONS

- NRR DATA NEEDS AND COMMUNICATION

- INTERFACE OF ON-SITE AND OFF-SITE

FURTHER LESSONS TO BE LEARNED, CONT.

- TRANSIENT AND ACCIDENT ANALYSIS

ROLES OF REALISTIC AND CONSERVATIVE ANALYSIS

ANALYSIS REQUIREMENTS FOR TRAINING

NRC CODE DEVELOPMENT

CODE VERIFICATION

APPENDIX E

RECOMMENDATIONS OF THE NUCLEAR POWER PLANT

EMERGENCY REVIEW PANEL

STATE OF CALIFORNIA

MAY 18, 1979

Memorandum

To : Edmund G. Brown Jr.
Governor

Date : May 18, 1979

Subject: Recommendations of
the Nuclear Power Plant
Emergency Review Panel

From : Russell L. Schweickart
Assistant for Science
and Technology

In response to your request, I am transmitting to you the recommendations of the Nuclear Power Plant Emergency Review Panel. These recommendations are designed to improve our emergency response capabilities to provide greater protection to the public in the event of a nuclear power plant accident here in California.

In arriving at these findings, representatives of the panel met with federal, state and local officials, consulted with utility representatives, and held public hearings in Los Angeles and Sacramento. Additionally, panel members reviewed many reports, studies, transcripts and other documents in their deliberations. I am confident, therefore, that we have learned from the Three Mile Island experience and that the actions which we propose will significantly improve our ability to deal with nuclear power plant accidents here in California.

Upon your concurrence and favorable action by the Legislature, adoption of the Panel's recommendations would have the following major results:

1. The area around California's nuclear power plants for which evacuation should be planned will be expanded to account for "Class 9" accidents (core meltdown and breach of containment). Thus a larger number of citizens, who live in the vicinity of nuclear power plants, will be covered by emergency plans and protective measures.
2. Timely notification of public authorities in the event of incidents with possible public health consequences would become certain and mandatory under state law. Additionally, critical occurrences at the plants, such as activation of emergency core cooling systems and high radiation levels would automatically send alarms to local and state emergency control centers.

Governor Brown
page two
May 18, 1979

3. Under those conditions at a nuclear power plant which may require emergency actions by state and/or local authorities to protect the public health and safety, a certified state official will be dispatched to the power plant to serve as the state liaison officer and on-site representative. This will assure credible communications between the plant and emergency response authorities.
4. The Department of Health Services will procure and deploy potassium iodide tablets to local emergency response agencies. These tablets, which protect the thyroid from radioactive iodine, will be distributed to the public endangered by an actual or imminent release of these materials in the event of a nuclear power plant accident.
5. Educational materials on radiation, its potential effects, and the kinds of events which can occur at nuclear plants will be distributed to the public within the emergency planning zone around nuclear power plants. Also included will be instructions on protective measures and emergency plans including evacuation information.

While it will require additional resources to improve our readiness to deal with possible nuclear power plant accidents, the panel has concluded that these efforts should be funded by the utilities operating nuclear power plants. The cost of emergency preparedness is an additional expense associated with this power source, hence, it is appropriate that those expenses be offset by the power companies and not the taxpayers in general.

It is my feeling that with implementation of these recommendations our ability to protect the public from the consequences of a serious nuclear power plant accident here in California will be considerably improved. It is essential that all of us recognize that no matter how diligent the nuclear regulatory authorities and the power plant operators are in assuring safety in the operation of these facilities, there will be serious nuclear accidents. We must, therefore, be prepared to act decisively in these circumstances to minimize the consequences to the California citizens whom we serve.

NUCLEAR POWER PLANT EMERGENCY REVIEW PANEL RECOMMENDATIONS

A. PLANNING AND EMERGENCY RESPONSE

Issue: Present planning for nuclear power plant incidents is based on the likelihood of a so-called "design basis" accident which might affect those living in the low population zone, 3-6 miles from the plant. The experience at Three Mile Island indicates that the magnitude of nuclear power plant accidents could be greater than previously believed and that the effects of such accidents could extend beyond the scope of existing emergency plans.

Recommendation A-1.

The California Energy Commission should determine the probable consequences of a release of radioactive materials from nuclear power plant accidents involving core melting and breach-of-containment (class 9 accidents). As part of this study, the Energy Commission should prepare, within six months, site specific maps for each of California's commercial nuclear power plants showing the areas likely to be affected by such a release. These areas shall be designated as Emergency Planning Zones, where evacuation or other protective actions would be required and where the food chain would be affected. This analysis would consider the size and type of reactor, local topography, weather conditions and other relevant factors.

Implementation: Governor's Directive

Recommendation A-2.

The State Office of Emergency Services (OES) should revise its Nuclear Power Plant Emergency Response Plan to reflect the information provided by the Energy Commission; a similar upgrading of other state agencies' plans and procedures should be ordered.

Implementation: Governor's Directive

Issue: Local authorities have the primary responsibility for preparing emergency response plans for nuclear power plant accidents. Like the state plan, however, local plans are based on the present planning criteria. Furthermore, many counties outside the existing low population zones have no plans, but are likely to be within the enlarged planning areas delineated by the California Energy Commission.

Recommendation A-3.

Local planning for nuclear power plant accidents should be mandated by state law for any jurisdiction where protective action might be required. County plans should be based on guidance developed by the State Office of Emergency Services, consistent with information provided by the Energy Commission. OES should continue to be responsible for coordination and approval of these plans.

In developing its approval criteria, OES should give special attention to problems of large-scale evacuations, appropriate in-place protective actions, as well as special protection for the handicapped and those in hospitals, convalescent homes, correctional facilities, etc. In addition, OES should ensure that adequate communication links and equipment are available at the local level.

Implementation: Legislation requiring state-mandated local program.

Issue: Current Nuclear Regulatory Commission (NRC) regulations do not adequately recognize the need for off-site emergency planning for nuclear power plant accidents. NRC does concur in state plans, but this process is not mandatory for licensing or operation of a nuclear facility.

Recommendation A-4.

The State should urge passage of new federal legislation which would prohibit the granting of a commercial nuclear power plant operating license until local emergency response plans are approved by state authorities and state plans have received concurrence from the NRC. The Energy Commission should examine its authority to impose this requirement on any future nuclear plants in California.

Implementation: Governor's request to the California Congressional Delegation and the NRC.

B. NOTIFICATION AND ASSESSMENT OF EMERGENCY CONDITIONS

Issue: Adequate emergency response to nuclear power plant accidents depends on timely and accurate notification of local and state authorities by the plant operator. Emergency notification procedures should be initiated before an accident has reached serious proportion, so that state and local officials can take preliminary steps to respond. At present there are no uniform criteria for notification and local authorities complain of uncertainty as to when and under what circumstances they will be alerted.

Recommendation B-1.

The State Office of Emergency Services with the Energy Commission, local authorities, and plant operators, should develop within 90 days a detailed, objective set of notification procedures. These should guarantee that plant operators provide timely notification to off-site authorities in the event of a nuclear power plant incident which threatens to have consequences beyond the plant boundaries. These criteria would require notification in events such as the activation of emergency systems within the plant, evacuation of plant personnel from certain areas, as well as incidents not directly related to the operation of the reactor, e.g., fires, sabotage, etc.

These procedures should recognize the need for the earliest practical notification of local and state authorities of situations which are potentially serious. Uniform application of these procedures should be mandatory under state law.

Implementation: Governor's Directive
Legislation

Recommendation B-2.

The State should require the installation of automated systems in each commercial nuclear power plant. These systems would trigger alarms in the OES State Warning

Center and at designated local agencies. The alarms would be activated by any of the following: initiation of emergency core cooling systems, indications of high radiation in the containment building and indications of excessive radiation levels present in the release of stack gases.

Threshold levels for these devices should be established by the ARB, the Energy Commission, and the Department of Health Services, Radiologic Health Section.

Implementation: Legislation

Issue: During a nuclear power reactor emergency, communications between the plant and off-site authorities are of critical importance. Decisions to evacuate or take other protective steps must be based on an accurate and timely flow of information. At present, state and local officials must rely on periodic reporting of critical events by the utility operator as the basis of emergency actions. At Three Mile Island, this dependency resulted in confusion, misinformation, and a lack of credibility which is unacceptable in arriving at sound decisions.

Recommendation B-3.

The State Office of Emergency Services should train and certify individuals who would be dispatched to a nuclear power plant site upon confirmation of any emergency having potential off-site consequences (e.g. activation of at least two of the warning alarms specified in Recommendation B-2). This official's sole responsibility would be to maintain liaison with state and local emergency officials, apprising them of developments at the plant and recommending appropriate response actions.

Since this proposal may involve areas of federal jurisdiction, the Governor should request that the Nuclear Regulatory Commission, as part of its accident response plans, approve the placement of such state officials at the plants during an emergency and ensure that appropriate communications capabilities are in place.

Implementation: Governor's Directive to OES
Governor's letter to the Nuclear Regulatory Commission

C. TRAINING AND PRACTICE EXERCISES

Issue: At present there are few training programs which can prepare state and local officials to fully carry out their responsibilities in the event of an accident. Counties not previously involved in nuclear emergency planning are particularly in need of adequate training programs, especially in radiological monitoring and assessment of health effects.

Recommendation C-1.

The California Specialized Training Institute (CSTI) should incorporate nuclear emergency response into its emergency planning courses for local and state officials. Further, CSTI and the Office of Emergency Services should develop a program of simulation exercises to present realistic crisis management problems to decision makers responsible for handling nuclear power plant accidents. These exercises should be conducted at least annually, on location where possible.

Implementation: Governor's Directive

Recommendation C-2.

The State Office of Emergency Services along with the Department of Health Services and health physicists employed by the utilities, should develop radiological protection training for local and state health officials. Certification of local officials by the Department of Health Services should be required as part of the approval process of local emergency plans.

Implementation: Governor's Directive
Legislation mandating local programs

D. MONITORING OF RADIOACTIVE RELEASES

Issue: Monitoring equipment adequate to measure total radiation dosage or dose rate may not exist at all necessary locations.

Recommendation D-1.

The Department of Health Services in conjunction with local officials should review the radiological monitoring requirements for the areas surrounding each nuclear power plant. Where necessary, low level radiation monitoring devices should be provided. The Department of Health Services Radiologic Health Section should develop a system for the rapid analysis of release products and total dose exposure. The Department of Health Services should make arrangements for the use of the computerized analysis capabilities (ARAC) at the Lawrence Livermore Laboratories for predicting the geographical distribution of radioactive materials released from a nuclear power plant. It should also arrange with university and private laboratories for analysis of the large number of samples which would be collected during an emergency.

Implementation: Governor's Directive

Recommendation D-2.

The Air Resources Board should set air quality standards for airborne radiological contaminants and establish source monitors within the boundaries of nuclear power plants to enforce these standards. These monitors would provide a continuous data readout and be connected to the existing ARB air quality monitoring system.

Implementation: ARB Order

E. PROTECTION OF PUBLIC HEALTH

Issue: Ingestion or inhalation of radioactive iodine, a potential release product from nuclear reactors, can result in an accumulation of these substances in human thyroid glands and cause serious damage. A thyroid blocking agent, potassium iodide, has been approved for distribution to the public; however, no stocks of this substance exist at present. Hence, the protection afforded by this drug is not available.

Recommendation E-1.

The Department of Health Services in conjunction with the Federal Food and Drug Administration should, within the next 6 months, develop a program for the procurement and distribution of potassium iodide tablets to emergency response agencies. This distribution should be accompanied by cautionary advice on the use of the substance and its proper storage. The method of ultimate distribution of these products to the public in an emergency should be specified in local emergency response plans.

Implementation: Governor's Directive

Recommendation E-2.

The Department of Health Services and the Office of Emergency Services should develop a plan for predistribution of potassium iodide to those persons for whom it is determined that evacuation could not be carried out prior to their exposure.

Implementation: Governor's Directive

Issue: Present emergency plans do not adequately address the problem of radiological contamination of the food chain (the "ingestion pathway"). New federal guidance calls for ingestion pathway monitoring up to 50 miles from a nuclear power plant. Other studies have suggested much larger areas might be affected. In any case local capabilities to monitor food chain contamination would be taxed.

Recommendation E-3.

The Department of Health Services should assume responsibility for ingestion pathway monitoring following a nuclear power plant accident. Health Services should procure in situ sampling and analysis equipment and plan to monitor up to 200 miles from a power plant site. Training and information on sampling and contamination control should go to state and local officials who would assist the Department in this task.

Note: The Health and Safety Code gives the Department of Health Services authority to impound any crop or food product which, in its judgment, is affected with harmful levels of contamination. The Department and appropriate federal authorities should jointly establish threshold levels which would initiate this action.

Implementation: Governor's Directive

Issue: Under existing criteria, only minimal involvement of medical facilities is anticipated, but, in the event of a large power plant accident, emergency medical facilities could be overwhelmed.

Recommendation E-4.

The Department of Health Services and the State Office of Emergency Services should assist local jurisdictions in identifying those hospitals and other care facilities within a 50-mile radius of each nuclear power plant capable of handling radiological and trauma injuries. They should also identify provisions for large scale decontamination, handling of exposure cases and long-term monitoring of health effects on those exposed to radiation.

Implementation: Governor's Directive

F. PUBLIC EDUCATION

Issue: The experience at Three Mile Island clearly points out the need to provide better information to the public before and during a crisis at a nuclear power plant. At present the general public has little understanding of the nature of nuclear accidents, the hazards of radiation or possible protective measures they might take.

Recommendation F-1.

Information concerning accidental releases from nuclear power plants should be distributed to the public in the vicinity of nuclear sites. The State Office of

Emergency Services, assisted by the Department of Health Services and the Energy Commission, should prepare the following information for distribution to the public:

- a. A brief description of radiation terms and dose comparisons including a description of relative health hazards resulting from various types of exposure.
- b. A brief explanation of the kinds of events that can occur as a result of a serious reactor accident.
- c. Information about countermeasures and protective actions which can be taken by the public.

This information should be distributed twice annually via utility bill inserts or a similar mechanism. Additionally, information on nuclear power plant accidents should be made a part of the "Survival Guide" which now appears in telephone directories in many areas of California.

Along with the general information described above, additional materials should be made available to those living in closest proximity to the plants (i.e., the Emergency Planning Zone as defined by the Energy Commission in Recommendation A-1). This would include information on the availability of potassium iodide as a thyroid blocking agent and special instructions concerning possible evacuation in the event of a reactor accident.

Implementation: Governor's Directive
PUC Order to private utilities
Legislation covering municipal utilities

G. LAND USE PLANNING

Issue: Current low density areas proximate to nuclear power plants face the prospect of significant population growth over time as the demands for housing increase. These demographic changes may erode the ability of local government to plan for the effective evacuation of at-risk populations in the event of an accident.

Recommendation G-1.

The Office of Planning and Research should amend the state guidelines for preparation and approval of Environmental Impact Reports under the California Environmental Quality Act. These guidelines should define within the class of "Mandatory Significant Effects" any proposed development or general plan change whose implementation may inhibit the execution of local emergency response planning for nuclear power plant accidents.

Environmental Impact Reports including such analysis should be reviewed for accuracy and adequacy by local and state offices of emergency services as well as the Energy Commission.

Implementation: OPR and Resources Agency

H. FINANCING FOR IMPROVED EMERGENCY PLANNING

Issue: Expanded emergency planning efforts to adequately prepare for nuclear power plant accidents will require increases in funding and staffing at the state and local level, beyond amounts currently budgeted.

Recommendation H-1.

The additional costs associated with improved emergency planning for nuclear power plant accidents should be borne by the utility operators themselves, since these expenses are part of the true costs of nuclear electric power generation in California. Therefore, the Office of Emergency Services, Department of Health Services, and the California Energy Commission should determine costs associated with development of emergency plans, training of personnel, procuring of equipment, and establishment of a public education program to meet the above outlined recommendations. The Public Utilities Commission and the Department of Finance should then establish an assessment structure whereby utility companies would reimburse state and local agencies for these expenses. The utilities should pay only those costs which can be assigned to nuclear preparedness, per se, as opposed to emergency planning in general.

Note: The costs of actually executing nuclear power plant response plans and the cleanup expenses following an accident may be substantial. Consideration of how these costs will be met was beyond the scope of this panel since that involves questions of liability insurance and the Federal Price-Anderson Act.

Implementation: Legislation

I. SUPPLEMENTAL RECOMMENDATIONS

Issue: While the federal government, through the Department of Energy and the Nuclear Regulatory Commission have responsibility for the handling of radioactive wastes, there are no current plans for the disposal of waste materials resulting from a nuclear power plant accident.

Recommendation I-1.

The State should insist that the Department of Energy and the NRC develop a plan for the handling and timely removal from the State of radioactive materials, including possibly portions of the plant itself, following a nuclear power plant accident.

Implementation: Governor's request to the NRC and Department of Energy

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