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

9/14/79

BILL—

ATTACHED ARE THE RESPONSES TO THE QUESTIONS WHICH RESULTED FROM THE APRIL 30 HEARING BEFORE THE SENATE SUBCOMMITTEE ON NUCLEAR REGULATION (MINUS THE ATTACHMENTS) AS YOU REQUESTED.

Spiros Droggitis
OCA
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QUESTION 1. It appears after the Three Mile Island accident that both the Commission in its regulatory program and the designers of commercial power reactors may have focused too much attention on the most severe types of events, which are also probably the most unlikely, and too little attention on less severe events which may be more likely. Would you agree with that assessment? If so, should NRC be reviewing its reactor regulation and research programs to determine whether priorities need to be adjusted?

ANSWER. (NRR)

The accident scenario at Three Mile Island Unit 2 can be summarized briefly as an anticipated transient that, through a combination of design shortcomings, equipment malfunction, and human error, turned into an only partially mitigated small-break loss-of-coolant accident. Both the anticipated transient (loss of normal feedwater) and small break loss-of-coolant accidents were evaluated in the course of NRC's licensing review for the Three Mile Island Plant. These evaluations did not include multiple equipment malfunctions and human errors that occurred at Three Mile Island.

In the past the licensing review has concentrated much attention on severe types of events, such as large-break loss-of-coolant accidents (LOCAs). Less resources (dollars and people) have been expended on the review of anticipated transients and small break LOCAs. The NRC staff will soon complete an initial assessment of the generic licensing implications of the Three Mile Island accident, and a review of the research program is also underway. Both of these reviews are sufficiently complete at this time to conclude that priorities need to be adjusted to give greater attention to anticipated transients, to small break LOCAs, and to the consideration of multiple equipment malfunctions or operator errors in the training of operations organizations. It is also likely that additional licensing review and research resources will be necessary in order to accomplish and implement the regulatory changes deriving from the Three Mile Island experience.

Research efforts presently under consideration include acceleration of experimental investigations of anticipated transients and small break LOCAs in the Semiscale and LOFT facilities, development of best-estimate computer codes, studies of fuel/coolant chemistry and fission product release and transport after fuel failure, as well as examination of the TMI core. Other research topics, including those related to improving reactor safety are discussed in the answer to question 48.

QUESTION 2. At what specific point during a severe transient such as that which occurred at Three Mile Island do NRC's regulations require that NRC, and State and local emergency preparedness officials be notified?

ANSWER. (IE/ELD)

The point in time for required notifications is dependent on the particular circumstances involved. 10 CFR Part 70 provides for immediate reporting for certain incidents involving licensed material. For example, 10 CFR 20.403 requires immediate reporting for incidents which cause or may threaten to cause certain exposures, releases, or loss of one or more weeks of operation or property damage in excess of $200,000. The report is due immediately after the evaluation is made by telephone. There is no time requirement placed on conducting the evaluation.

Additional NRC reporting requirements applicable to events which occur at licensed plants are contained in each facility's Technical Specifications. The Technical Specifications require prompt notification with a written followup report for various nonroutine events. The definition of prompt is as expeditiously as possible, but within 24 hours.

State and local notifications are covered in the facility Emergency Plans that are mandatory and usually submitted with the licensee's Final Safety Evaluation Report. The plans are site-unique and result from negotiations between the licensee and the State and local authorities. The notifications are usually based on events which result in offsite release of radioactive material or that a potential for release exists.

QUESTION 3. Much attention has been given to the fact that NRC was not notified by the licensee until nearly four hours after the start of the transient. It appears that State and local officials were notified at about the same time. Sho ld NRC, State, and local officials have been notified earlier by the licensee that the transient was occurring?

ANSWER. (IE)

In our view, the licensee should have notified both the NRC and State and local officials earlier. Based on the TMI experience, we plan to revise NRC's 10 CFR 50, Appendix E, Emergency Plans for Production and Utilization Facilities, and to Regulatory Guide 1.101, Emergency Planning For Nuclear Power Plants, to include more specific guidance regarding timely notification of the NRC and State and local officials.

QUESTION . At what point should NRC have been notified in your view?

ANSWER. (IE)

In our view, the NRC should have been notified within approximately one hour after it became apparent that conditions were not returning to a controlled or expected condition.

QUESTION 4. Based upon your knowledge at this time, did the licensee's notifications to you and to State and local officials meet the requirements of NRC's regulations? If so, does this indicate deficiencies in this aspect of the Commission's regulations? Why shouldn't NRC be notified promptly whenever any transient occurs?

ANSWER. (IE/ELD)

The licensee notified the Commission and State officials promptly after it declared a site emergency, at about 7:00 a.m. on March 28, 1979, approximately 3 hours after the feedwater pumps tripped. To the best of our present knowledge, this was consistent with the reporting requirements of 10 CFR Part 20 and the licensee's technical specifications. By practice, the NRC has considered a report within one hour of determining reportability to be sufficiently timely to meet the prompt reporting requirement.

We are examining our reporting requirements as a result of the TMI-2 incident. We believe that earlier notification is needed.

We are still examining the need for reporting transients. Many transients are of little or no safety consequence. Our objective has been to require reporting of significant safety information promptly, and we have tried to define significant safety information to get prompt reports of essentially all such information. Since the TMI-2 event, we have issued bulletins to all licensees of operating nuclear reactors which included the following action to be taken by the licensees:

> Review your prompt reporting procedures for NRC notification to assure that NRC is notified within one hour of the time the reactor is not in a controlled or expected condition of operation. Further, at that time an open continuous communication channel shall be established and maintained with NRC.

We believe this type of communication will enhance our ability to respond to and assess plant conditions.

QUESTION 5. There has been considerable criticism since the accident of NRC's initial response to the situation at the Three Mile Island nuclear plant, focusing particularly on NRC's failure to get a team of high-level reactor safety experts to the site within the first two days of the accident, and the apparent communications breakdown between the site and NRC headquarters. How would you rate your own performance in this situation, and what lessons have you learned about NRC's ability to react to such situations in the future? Why shouldn't NRC prepare in advance a detailed plan for getting qualified personnel with the authority to act for the agency to the scene of an emergency promptly, when they are needed? What specific elements do you believe should be contained in such a plan?

ANSWER. (IE)

At the time of the Three Mile Island incident, the NRC had developed and implemented an incident response capability for prompt dispatch, as appropriate, of a team of reactor systems and health physics inspectors from the NRC regional office to the site. Transportation provisions included automobile and chartered aircraft -- both fixed wing and helicopters.

Two hours and twenty minutes after notification to the NRC of the Three Mile Island incident, one reactor inspector and four health physicists arrived at the site. A total of eleven NRC inspectors were on site the first day of the event. Seven analysts and engineers from NRC Headquarters in Washington, D. C. and sixteen NRC inspectors were on site on March 29. On March 30, two days after the initial event, the NRC had approximately 85 personnel at the site. The NRC recognizes in retrospect that more rapid deployment of personnel with a 24-hour staffing capability is necessary for an emergency like Three Mile Island. The resident inspection program will reduce the time required to get a highly trained plant systems inspector to the site for the initial response since the inspector will be living in the site vicinity.

Preplanning for accidents did not include prompt dispatch of high-level NRC management to the accident site from NRC headquarters. Rather, preplanning envisioned that the accident would unfold so rapidly that high level NRC management would be precluded, due to time, from having on-site influence on the response to the incident. This concept of a high-level NRC management team promptly onsite in the event of a major accident will be included in the future work of the task force on lessons learned from the TMI accident.

ANSWER (Con't)

Thus, the NRC believes the additional pre-planning for getting a highly qualified team of personnel, including high level NRC management, from both the regional offices and NRC Headquarters promptly to the scene of an emergency is warranted. The plan should include the following elements:

Implementation Criteria
Mission
Authority
Team Organization
Team Composition
Transportation Provisions

QUESTION 6. Questions have been raised about NRC's ability to get the licensee in an emergency situation to follow a particular course of action in dealing with the accident. What authority does the Commission have to require the licensee to follow a particular course of action? Under what circumstances do you believe that NRC personnel should substitute their judgment for that of the licensed reactor operators on the appropriate action to take in an emergency situation? Should this only be done when NRC personnel are on-site and have first-hand knowledge of the situation?

In your view, is NRC's present legal authority sufficient to permit whatever action in this regard is needed? If not, what specific changes are needed? To what degree would action such as this expose the Federal government to liability for the further consequences of an accident?

ANSWER. (ELD)

The Commission regards its general supervisory authority under the Atomic Energy Act as sufficient to enable us to require NRC approval of all significant decisions made during the recovery and cleanup period following an accident at a power reactor. In particular, Section 161b. of the Act authorizes the Commission to "establish...by order, such standards and instructions to govern the possession and use of special nuclear material, source material, and byproduct material as the Commission may deem necessary or desirable...to protect health or to minimize danger to life and property." We view this provision as giving the Commission ample power to order a licensee to obtain Commission approval prior to any action significantly involving radiation health and safety.

Although the situation has not arisen at Three Mile Island, circumstances are conceivable in which prompt construction and operation of new equipment not covered by the facility license, newly designed decontamination facilities for example, might be necessary to reduce health hazards and assure continued safe shutdown at a damaged facility. In such emergency circumstances the Commission believes it has implied authority to order, effective immediately, that equipment be built and operated; and may conduct license amendment proceedings while, and after, the ordered action takes place. However, clarifying legislation would be desirable on this point. Therefore, the Commission believes that it would be desirable if Section 189 were to be amended to state explicitly that the Commission may issue an immediately effective license amendment upon a finding that public health and safety or common defense and security so requires. Such an amendment would not interfere with an interested person's right to request a hearing at which the Commission's action would be reviewed at a later time.

If the NRC were to take possession of and to operate a nuclear power plant in an emergency situation, the Federal government would be exposed to liability. How, and to what extent, is not clear, but certainly to the extent that the sovereign has consented to be sued under applicable law.

QUESTION 7. Did you receive full cooperation at all times from the licensee of the Three Mile Island plant in dealing with the emergency situation there? If not, please elaborate.

ANSWER. (NRR)

The cooperation by the licensee varied; it was better after the large contingent of licensing experts under Mr. Denton's direction arrived at the site than in the early hours following the accident. It now appears that in the early hours there may have been important information not brought to NRC's attention.

Early on there was also some lack of coordination of public statements by the licensee and the NRC, and there were differences in our respective understanding of current information. The licensee cooperated in the sense of responding to NRC directions throughout the conduct of accident recovery operations. After the NRC headquarters people arrived at the site, communications were much easier and a spirit of mutual cooperation soon developed. NRC was involved at all levels of the licensee's management of the accident recovery. There was a need to augment the licensee's technical and managerial resources at the site in the early days following the accident, especially to support the around-the-clock accident mitigation activities then underway. The licensee cooperated with NRC in solving this problem in a timely way.

QUESTION 8. Throughout much of the first five days of the accident at Three Mile Island, there appeared to be severe difficulties in obtaining accurate information about the events at the plant. A particular problem appeared to be conflicting reports on the situation from the licensee, NRC officials at the site, and NRC officials at headquarters. What was the source of these information difficulties and how can they be avoided in any similar situations in the future. Would it help to have a direct and reliable communication link between the control room, NRC, and appropriate State officials, together with the designation of one individual in the control room whose sole responsibility in the event of an emergency is to provide information to NRC and the State?

ANSWER. (IE)

Both the Region I Incident Response Center (IRC) and Headquarters IRC were activated promptly after the accident was reported. Concurrent with this activation, communications were established between the Region I IRC and Headquarters IRC. For a short period of time, between about 8:00 a.m. and 10:30 a.m., the Regional IRC relayed information between Headquarters IRC and the site. This procedure, as well as the time spent in clarifying information, resulted in delays in transmitting information. At about 10:30 a.m., communications was established between Region I personnel at the site and Headquarters IRC. Later on the 1st day of the accident, about 1:30 p.m., a conference call was established between the Headquarters IRC, Regional IRC and the site. This communications channel essentially remained open for the rest of the incident. Periodically, during the first 5 days following the accident, communications were broken for short periods and sometimes hampered by a high noise level.

Part of the communications difficulties stemmed from a lack of a clear definition of what type of information was needed and from a need to identify specific channels for reporting of information both to NRC personnel onsite and offsite. Following the arrival of Harold Denton, Director, Office of Nuclear Reactor Regulation, and other NRC high management personnel onsite on Friday, March 30, the public and Commission placed more confidence in the information reported from the site. The concept of promptly dispatching high level NRC management to a site in the event of an accident and its relationship to the communication of event information will be included in the NRC analysis of lessons learned from the TMI accident. This concept is discussed in further detail in the response to Question 5.

At our current stage of understanding of the TMI accident, we believe there may be some confusion between the ability to communicate as a mechanical situation and the ability to communicate promptly with accurate information. As we inquire further into this matter we will be evaluating our communication needs in greater detail, particularly with respect to the most efficient means of transmitting information between the facility, NRC and appropriate State officials.

QUESTION 8 (continued)

ANSWER

In an immediate effort toward improved early communications, the NRC has requested licensees to notify the NRC Regional Offices within an hour if a transient occurs which is not being controlled. The telephone used for this type of notification should be left open. To facilitate the requested improvements, we have had direct and dedicated telephone lines installed in the control room, reactor supervisor's office and other locations at all operating nuclear power plants. These telephones automatically ring at the NRC Headquarters Operations Center when the receiver is lifted off the telephone cradle. This system became operational June 1, 1979.

We believe that significant improvements can be made in obtaining and providing information to government officials and the public. We believe that there should be an individual in the control room who is solely responsible for providing information to NRC. We have already issued a Bulletin to all licensees which requires licensees in the event of an incident to notify the NRC within one hour and to open and maintain a continuous communication channel with the NRC.

QUESTION 9. Please describe NRC's role in the training and licensing of reactor operators. To what extent does that training and licensing program involve the use of plant simulators? To what extent are operators trained and tested in emergency situations such as those which occurred at Three Mile Island?

ANSWER - (NRR)

The NRC provides guidance to utilities regarding the training of operators in Regulatory Guide 1.8, "Personnel Selection and Training". Using this guidance and that provided in Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports" the utilities develop detailed training progr. These programs are reviewed by NRR pursuant to Standard Review Plan, Section 13.2, Training. Programs are approved when they meet the requirements in these documents, copies of which are attached.

Individuals who manipulate reactor controls must receive operator licenses. Individuals who direct the licensed activities of licensed operators must receive senior operator licenses.

10 CFR Part 55, "Operator's Licenses" describes the procedures and criteria for issuance of licenses to operators and senior operators. NUREG-0094, "NRC Operator Licensing Guide" further describes the licensing program. Copies of these documents are attached.

Training programs for operator and senior operator licenses vary depending upon whether the applicant will be licensed prior to or after initial operation of the facility.

The training programs described below for applicants to be licensed prior to initial operation are for individuals with no previous nuclear experience. Training programs for individuals with nuclear experience are modified as appropriate to take into account the prior experience.

In the first phase of training, the applicants are introduced to (a) the nuclear and chemical processes that occur in an operating reactor, (b) radiation and its effects, and (c) the necessity of operating a reactor in a responsible manner. The programs last for 12 weeks and conclude with each applicant participating in a 1-week laboratory course at a research reactor. This training includes operation of the research reactor.

In the second phase, the applicant attends a design lecture series where he learns the generic product lines and operating characteristics of the type of facility he will operate. This program lasts 6 weeks.

ANSWER (Question 9 continued)

In the third phase, the applicant operates the controls of a nuclear power plant simulator during normal, abnormal, and emergency conditions. As part of this training, the applicant resides at an operating power plant to observe day-to-day plant operations beyond those that can be taught in the simulated control room. This part of the program lasts 4-1/2 months. At the conclusion of the course, the applicant must successfully complete a written examination and an operating test similar to the NRC examination he must later pass.

In the final phase of training, the applicant returns to his facility to attend classes on the design features of the facility, write operating procedures, perform construction check outs and run preoperational tests of equipment. This phase lasts approximately 1 year. Just prior to taking an NRC examination, the applicant returns to the simulator training center for a 1-week refresher course.

Individuals who apply for licenses after the facility has achieved criticality normally receive all of their training at the facility where they will work. The programs are similar in scope to the programs for the precritical applicants. They include 3 months of control room experience. Individuals who participate in preoperational testing and startup testing do not normally attend a simulator course, although some may attend a 1- or 2-week simulator course. Most of these individuals have been at the plant for 3 or 4 years going through the normal job progression prior to sitting for the NRC examination.

During the preoperational test phase, the facility is nearing completion of construction, NRC inspectors confirm that the licensee has implemented the approved initial operator training program.

An individual who applies for an operator's license must pass a seven part written examination that is designed to be completed in eight hours. He also is required to pass an operating oral examination at his facility. This test requires between four and five hours to complete. Some facilities have simulators located on, or near, the site. We have used these simulators for examinations for applicants from these facilities.

A senior operator applicant, in addition to the above examinations, is required to sit for a five part written examination that is designed to last five hours. This examination requires a greater depth of knowledge and understanding of reactor theory, operating characteristics and license provisions than does the operator's examination. The oral test also requires a better understanding than the operator's test.

In each of the written examinations, questions on safety related systems, their operation and operator response to abnormal and emergency situations make up thirty to forty percent of the examinations. Twenty to twenty-five percent of the test is devoted to questions relating to operator response to abnormal and emergency situations.

ANSWER (Question 9 continued)

During the training programs, the applicants are impressed with the need to use and adhere to written procedures, for normal, abnormal, and emergency operations. The training programs, however, are also designed so that the individuals become intimately familiar with their plant and its operation so that they may reason their way through various transient situations and take appropriate action while remaining within the boundaries of the operating procedures and other administrative directives.

However, there are apparent weak areas in the training programs. A thorough review is being undertaken of the programs conducted at simulator training facilities.

In the present training programs, when the simulator is initialized for a particular training demonstration, all systems, valves, pumps, etc., are in the correct position for that mode when the student enters the scene. The student is not required to verify or realign the various components.

During training exercises, students observe various malfunctions to equipment and transient conditions that result in the actuation of emergency cooling systems. During simulator examinations, the student is expected to recognize these events and take appropriate action. However, when emergency systems are actuated, they always work; no malfunctions in emergency systems are presently programmed. Therefore, the student walks through a procedure "verifying" that the automatic action has taken place. He never has to open a valve, start a pump, etc., in an emergency system.

The Commission presently has under consideration recommendations to revise nuclear power plant simulator standards so that malfunctions are programmed for emergency systems.

Q 9

PART 55

OPERATORS' LICENSES

AUTHORITY: The provisions of this Part 55 issued under secs. 107, 161, 68 Stat. 939, 948; 42 U.S.C. 2137, 2201. For the purposes of sec. 223, 68 Stat. 958, as amended, 42 U.S.C. 2273, §55.3 issued under sec. 161i., 68 Stat. 949; 42 U.S.C. 2201 (i). Sec. 55.40 issued under secs. 186, 187, 68 Stat. 955; 42 U.S.C. 2236, 2237. Secs. 202, 206, Publ. L. 93-438, 88 Stat. 1244, 1246; 42 U.S.C. 5842, 5846.

GENERAL PROVISIONS

§ 55.1 Purpose.

28 FR 3197

The regulations in this part establish procedures and criteria for the issuance of licenses to operators, including senior operators, of facilities licensed pursuant to the Atomic Energy Act of 1954, as amended, or section 202 of the Energy Reorganization Act of 1974 and Part 50 of this chapter; and provide for the terms and conditions upon which the Commission will issue these licenses.

§ 55.2 Scope.

40 FR 8774

The regulations contained in this part apply to any individual who manipulates the controls of any facility licensed pursuant to Part 50 of this chapter and to any individual designated by a facility licensee to be responsible for directing the licensed activities of licensed operators.

§ 55.3 License Requirements.

28 FR 3197

(a) No person may perform the function of an operator as defined in this part except as authorized by a license issued by the Commission;

(b) No person may perform the function of a senior operator as defined in this part except as authorized by a license issued by the Commission.

§ 55.4 Definitions.

As used in this part:

(a) "Act" means the Atomic Energy Act of 1954 including any amendments thereto.

(b) "Commission" means the Nuclear Regulatory Commission or its duly authorized representatives.

(c) "Facility" means any production facility or utilization facility as defined in Part 50 of this chapter.

(d) "Operator" is any individual who manipulates a control of a facility. An individual is deemed to manipulate a control if he directs another to manipulate a control.

(e) "Senior operator" is any individual designated by a facility licensee under Part 50 of this chapter to direct the licensed activities of licensed operators.

(f) "Controls" when used with respect to a nuclear reactor means apparatus and mechanisms the manipulation of which

Attachment 4
Q 7.

U.S. DEPARTMENT OF COMMERCE
National Technical Information Service

PB-255 664

A Guide for the Licensing of Facility Operators Including Senior Operators

Nuclear Regulatory Commission

July 1976

QUESTION 10. What retraining and relicensing requirements are now imposed by NRC on reactor operators? What measures are being taken by NRC and by those who train operators to revise the training program to focus more directly on operator response to emergency situations? What measures should be taken? Is the requirement for periodic retraining of operators on simulators warranted?

ANSWER. (NRR)

Operators and senior operators must renew their licenses every two years. The licenses are renewed without NRC reexamination, provided the Commission finds that: (1) the individual has been actively engaged under his license; (2) continues to meet the physical requirements and (3) has successfully participated in an NRC approved requalification program.

The requalification programs consist of annual written examinations, oral tests, lecture series, minimum number of control manipulations to be performed and review of procedure, design and license changes. The programs are administered by the licensee and audited by NRC. During the operational life of the facility, Inspectors from the NRC Office of Inspection and Enforcement confirm, on an annual basis, that the licensee's requalification training program meets the requirements of the approved program. Requalification training course content and examination content are audited by the Operator Licensing Branch of the Office of Nuclear Reactor Regulation.

Requalification programs permit operators and senior operators to execute all of their control manipulations at the facility. The vast majority of these are normal manipulations. Therefore, the majority of the operators walk through abnormal and emergency procedures. Consideration is being given to requiring all operators to attend simulators as part of the requalification programs.

The Commission presently has under consideration staff recommendations that would require all licensees to attend simulators representative of their facility annually. In addition, NRC would specify the exercises to be performed by each licensee. Finally, NRC examiners would administer some requalification examinations using the simulators to assure the validity of the programs.

QUESTION 11. It appears that one significant deficiency at Three Mile Island was the lack of detailed engineering back-up for the plant operators to deal with the emergency situation. Would you agree? Shouldn't each utility be required to develop a plan in advance to assure that hte necessary engineering back-up is available when needed? Shouldn't NRC be required to certify that plan and to develop criteria for certification?

ANSWER. (NRR/ELD)

On the basis of the information available to date, it appears that additional training or other preparation of the operating staff and the Metropolitan Edison engineering/management staff might have led to earlier corrective actions to mitigate the consequences of the accident within the first several hours of the initiating transient.

Further information has been developed on this matter by the investigation of the accident by the NRC Office of Inspection and Enforcement (see Attachment 1 to Q45), including detailed examination of computer printouts and other instrument records, and interviews with the operations and management staff at Three Mile Island. From the results of this investigation, it appears that this preliminary judgmenet is correct. Several possible remedies are certainly worthy of further consideration, including the development of licensing requirements for engineering staff training and planning for emergency support of reactor operators. This sort of requirement would need to be closely coordinated with improved operator training requirements discussed in response to questions 9 and 10.

A recent report, "TMI-2 Lessons Learned Task Force: Status Report and Short-Term Recommendations," NUREG-0578 (see Attachment 2 to Q45), contains several recommendations bearing on engineering back-up for emergency situations. In addition to recommendations which would clearly define the Shift Supervisors responsibilities for command and control of plant operations during routine and emergency operations, the Task Force recommended that a Technical Advisor be added to each shift. The Shift Technical Advisor would be required to have an engineering degree or equivalent, intimate knowledge of the characteristics of the plant and be thoroughly knowledgeable in its operation. While he would not have operational responsibilities, he would be immediately available to provide advice to the Shift Supervisor in the event of an emergency. Flexibility will be maintained in implementing this recommendation to accommodate differences in organization and manning at various utilities, but the objective of the recommendation will be adhered to.

One additional aspect of the concept of improved engineering backup for the plant operators is the need to require licensees to establish an on-site technical support center that could be promptly manned by engineering and management for close support of control room operations and direct access to NRC so as to improve overall management control of emergency situations and coordination with State and Federal Authorities. The Task Force has recommended that such centers be established at all power reactors, in close proximity to the control room, and that the centers be equipped with display and plant status monitoring capability necessary for engineering support. The centers would provide a key communications link to NRC. Work is underway on evaluating what specific kinds of support are needed and on the planning and preparation necessary to assure it will be available when necessary.

QUESTION 12. This Subcommittee has received testimony from a previous witness that events which occurred at the Davis-Besse and Rancho Seco plants were advance warnings of what later occurred at Three Mile Island. Do you believe this to be the case? If not, why not? What actions did the NRC staff take in response to the Davis-Besse and Rancho Seco incidents?

QUESTION. Do you believe this to be the case?

ANSWER. (NRR)

Over the years there have been numerous transients in pressurized and boiling water reactors and plant response has been favorable for most of these. The Davis-Besse and Rancho Seco transients were among the most severe. Two feedwater type events were reported to have occurred in 1977 at the Davis-Besse 1 facility (i.e., September 24th and November 29th). The September 24th event was similar to the early stage of the TMI-2 accident in that there was a feedwater system malfunction that caused the pressure in the primary system to rise and actuated the pressurizer power operator relief valve (PORV). The system depressurized because the PORV failed to close; however, the operator diagnosed the problem and closed the block valve thereby terminating the blowdown. Subsequent operator action included the use of makeup pumps and high pressure injection to bring the plant to a cold shutdown condition. While similar to the TMI-2 accident, this event was different in that the initial power level was 9% vs. 97% and there was auxiliary feedwater available to one steam generator.

The November 29th event resulted from a loss of offsite power leading to system depressurization and subsequent natural circulation cooling. This event was not really a TMI-2 accident type scenario. Of special significance during this event was a reduction of pressurizer water level due to primary coolant volume decrease resulting from cooldown.

Two feedwater type transients occurred at the Rancho Seco facility (i.e., March 20, 1978 and January 5, 1979). These events had similar initial behavioral characteristics to the TMI-2 as to feedwater system involvement and reactor pressurization and depressurization, but in both cases, the events were terminated before any core damage occurred.

The March 20, 1978 event that led to a rapid cooldown occurred as a loss of non-nuclear instruments including steam generator, pressurizer level and all reactor coolant temperatures. This caused an interaction with the Integrated Control System (ICS) which led to a loss of feedwater flow to the steam generators. With a loss of heat removal, the reactor pressure increased to a high pressure reactor trip. Auxiliary feedwater and possibly main feedwater flow to one steam generator started and rapid cooldown occurred. Reactor water inventory was maintained by the high pressure injection (HPI) pump.

The January 5, 1979 incident resulted from an electrical short in the ICS causing a reduction in feedwater flow leading to a pressure increase in the reactor coolant system and a high pressure trip occurred. Subsequent depressurization led to actuation of the HPI. The event was terminated after about 5 minutes by operator action.

The foregoing incidents were examined by members of the staff, analyses were provided, and procedures were changed at these plants as a result of these events. In retrospect, the transients should have been examined more generically and critically. If the analyses had considered additional failures, which they did not, the significance would have been clearer. In light of this, these events were not recognized as advance warnings.

QUESTION. If not, why not?

ANSWER. (NRR)

See above.

QUESTION. What actions did the NRC staff take in response to the Davis-Besse and Rancho Seco incidents?

ANSWER. (NRR/ELD)

The November 29, 1977 event at Davis-Besse was viewed by the staff with differing degrees of concern. Following the event, members of our inspection staff raised the matter of an unreviewed safety question because of the inability to follow the coolant inventory on plant instrumentation, due to insufficient range. There was concern also with the volume of the pressurizer. Thus, the plant was kept from returning to power until an analysis was provided by the licensee. The analysis was provided and reviewed and discussed by the inspection and licensing staffs. They concluded that the event did not constitute an unreviewed safety question.

An inspector continued to have reservations about the ability of the plant to sustain feedwater transients. In January 1979, he asked that this concern, along with several others, be provided to the Licensing Boards for Davis-Besse Units 2 and 3 and Midland Units 1 and 2 for their consideration. This information was forwarded in late March 1979.

The information provided by the inspector for the Boards was independently evaluated and that evaluation recognized that some transients result in pressure and volume changes, as he believed, that are beyond the ability of the pressurizer and normal reactor coolant makeup systems to control, but concluded that they could be sustained without compromising the safety of the reactor. These analyses did not consider exacerbation by lack of timely addition of feedwater to steam generators, loss of circulation, and other factors experienced at TMI-2. Had these factors been considered, the conclusions should have been that some safety features as provided in IE Bulletins subsequent to TMI-2 would be required. The pressurizer level problem was identified for further review.

The Rancho Seco event was severe but the safety concern was different. On March 20, 1978, loss of non-nuclear instrumentation caused termination of feedwater flow. Subsequently, the reactor coolant system experienced a rapid cooldown. Although the system pressure decreased, it did not reach the corresponding saturation temperature. Hence, no voiding occurred. The event review focused primarily on the effects of rapid cooldown and the consequences related to vessel and piping stresses. The licensee was asked to provide an analysis of the circumstances and the operations schedule was delayed until it was performed. The analysis was reviewed by the inspection, and in part, by the licensing staffs. The review resulted in some changes to procedures relating to instrumentation needs, and the plant schedule was resumed.

The staff has completed a preliminary review of feedwater type transients in B&W plants. The results of this review are reported in NUREG-0560 (attached). One of the findings made is that a study should be conducted by NRC of the entire reporting and data-assembly processes followed to accumulate and assess the significance of operating plant data. The specific objective is to be able to identify at an early stage those events which have a high recurrence frequency that challenge the safety systems.

All licensees receive copies of Licensee Event Report Summaries and regular "current events" reports prepared by the NRC staff. These would include the Davis-Besse and Rancho Seco events. However, in the past they would not have emphasized aspects of generic concern related to such events. That aspect must receive greater attention. As a corollary, the Lessons Learned Task Force has emphasized the need for licensee personnel to systematically review Licensee Event Reports which might affect their own plants (See NUREG-0578-Attachment 2 to Q45).

ATTACHMENT

NUREG-0560

Q 12

STAFF REPORT ON THE GENERIC ASSESSMENT OF FEEDWATER TRANSIENTS IN PRESSURIZED WATER REACTORS DESIGNED BY THE BABCOCK & WILCOX COMPANY

UNITED STATES NUCLEAR REGULATORY COMMISSION

Office of Nuclear Reactor Regulation

U. S. Nuclear Regulatory Commission

QUESTION 13. The Advisory Committee on Reactor Safeguards has provided this Subcommittee with a technical report dated January 1978 from C. Michelson, a TVA employee, entitled "Decay Heat Removal During a Very Small Break LOCA for a B&W 205-Fuel Assembly PWR". That report appears to raise a number of concerns regarding the ability of the B&W plant safety systems to respond to a small break LOCA and the adequacy of emergency procedures and operator training to deal with a small break LOCA. Apparently, the report was submitted to B&W in 1978. When did the NRC staff receive a copy of this report? Is this report something which should have been forwarded to the NRC staff earlier? What is the significance of the report's findings for the operating safety of B&W plants under conditions such as those which occurred at Three Mile Island?

QUESTION. When did the NRC staff receive a copy of this report?

ANSWER. (NRR)

As far as we can ascertain at this time, the NRC staff first received a copy of the January 1978 Michelson report early in April, 1979. The report was not formally transmitted from TVA but informally provided by Mr. Michelson upon request of the staff. More recently, it has been learned that a copy of the Michelson report was formally transmitted to B&W by TVA in April 1978, and copies apparently were available to the ACRS. This action is presently being investigated by the NRC's Office of Inspector and Auditor. It is clear however, that the January 1978 report had not received formal NRC staff review prior to the TMI-2 accident.

We have also ascertained within the past few days that two handwritten documents which were apparently drafts of the material which eventually became the January 1978 Michelson report were informally provided to a member of the NRC staff in late 1977 or early 1978 by Jessee Ebersole, Mr. Michelson's supervisor at TVA and a member of the ACRS (copy enclosed). The staff member recalls discussing the general areas of natural circulation and the effects of noncondensible gases in about that time frame with Mr. Ebersole. He does not recall responding formally to the handwritten material provided by Mr. Ebersole. In January 1978, that same staff member originated a Reactor Systems Branch review reminder (copy enclosed) which in part treats the concerns raised in Mr. Michelson's report of January, 1978.

QUESTION. Is this report something which should have been forwarded to the NRC staff earlier?

ANSWER. (NRR)

From our review of the letters between TVA and B&W now available to us, it appears that the Michelson report was not considered by TVA to identify any specific safety problems, but rather to identify a number of concerns regarding core cooling during very small break LOCAs. Exchange of technical information, including concerns such as in the Michelson report, is frequently carried out between the vendors and the customers without NRC involvement. If the concerns identified in the January 1978 Michelson report were subsequently determined by B&W or TVA to involve defects which could create a substantial hazard, then they should have reported them to NRC.

Since TVA apparently did not know initially if any safety problem existed, and B&W, in a letter to TVA on January 27, 1979, subsequently asserted that none existed,* neither organization apparently believed it was necessary to report these findings to NRC since such reporting did not occur. However, NRC's Office of Inspector and Auditor is conducting an independent investigation in order to determine if failure to report the Michelson conclusions to NRC by either organization constituted a violation of the requirements of 10 CFR 21 (Reporting of Defects and Noncompliance).

QUESTION. What is the significance of the report's findings for the operating safety of B&W plants under conditions such as those which occurred at Three Mile Island?

ANSWER. (NRR)

The January 1978 report by C. Michelson documented concerns regarding the ability of the core to remain covered for breaks in the B&W primary coolant system smaller than those breaks normally analyzed for licensing applications. The concerns primarily focused on the lack of documented information which confirmed that the consequences of breaks presently considered for licensing applications conservatively bound the consequences of smaller breaks.

The basis for Michelson's concerns were hand-calculated steady-state mass and energy balances. It did not account for detailed effects (e.g., transient terms and geometry) usually modelled in the more sophisticated computer codes used for licensing.

The four most significant items of the report which had a direct bearing on the events at TMI-2 as well as the behavior of B&W plants to small breaks are as follows:

* In a more recent submittal; letter, J. H. Taylor to R. J. Mattson, dated May 7, 1979 transmitting "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant" (Volumes I and II), B&W confirmed the conclusions of their January 27, 1979 letter with more detailed evaluations and analyses.

1. Very small break plant response differs significantly from plant response to small breaks described in Safety Analysis Reports.

2. Natural circulation plays an important role in core cooling following very small breaks and could be interrupted because of steam formation in hot leg piping.

3. Pressurizer level indication is not a correct indication of system water inventory, and

4. Small break isolation by operator action causing system repressurization with subsequent relief and/or safety valve failure.

The report brought attention to the fact that very small breaks in the primary coolant system behave differently than small breaks previously analyzed and therefore provided an indication that different emergency procedures might be needed for very small breaks. The January response by B&W to the Michelson report did not address this question and no changes were made in the emergency operating procedures. The May 1979 submittal confirmed the behavior of the plants to very small breaks as described by Michelson and provided guidelines for the preparation of emergency procedures in the event of very small breaks. These guidelines are presently being adapted as emergency procedures for the various operating B&W plants.

The report also brought attention to the importance of natural circulation for very small breaks. From this, coupled with the thermal-hydraulic behavior of the Three Mile Island plant, it was learned that previous modeling representations were not sufficient, and that additional nodalization in the pressurizer and steam generator models was needed to more accurately represent the expected system behavior.

As a result of these model changes, analyses have confirmed Michelson's prediction that natural circulation could be interrupted. However, these analyses also showed that core uncovery does not occur for any of these very small breaks.

In the TMI-2 accident, the pressurizer level indicated the pressurizer was full of liquid. The operators mistakenly interpreted that to mean the system was full of water, and shut off the high pressure ECC injection. The indication of a full pressurizer while other parts of the primary system may be voided could also occur for small breaks analyzed for licensing. As stated in the enclosure, additional operating procedures will be given to all plant operators such that system pressure will be a main measurement system inventory determination. In addition, hot and cold coolant loop temperatures would also be used.

no!

In the TMI-2 accident the power-operated relief valve on the pressurizer failed to open during overpressure. Subsequent isolation of this failed valve with an upstream block valve resulted in break isolation.

While the events in the TMI-2 accident did not follow the sequence postulated by Michelson, both valve failure and "break" isolation did occur. The isolation of a break* is not specifically considered in safety analyses. B&W stated that the isolation of a small break and subsequent repressurization does not produce a less safe condition than not isclating a break. That is because any repressurization that results in relief and/or safety valve opening or failure is bounded by small break analyses with break sizes slightly larger than the valve opening size. The staff agrees in principle with this explanation. However, we will require all applicants and licensees to analyze very small breaks which exhibit repressurization with subsequent pressurizer valve failure as part of their evaluation of plant response to small breaks.

The significance of the Michelson report findings to conclusions regarding the acceptability of consequences due to small breaks in B&W plants will be addressed in detail in a staff report to be issued shortly. This is one of the considerations requiring resolution before restart of the presently shutdown B&W plants.

The key conclusions of the staff evaluation of the January 1978 Michelson report and the B&W response to the report are as follows:

1. The overall behavior of the plants to small breaks was shown to be, for the most part, consistent with the behavior as predicted by Michelson and, within the expected accuracy the B&W analyses substantiate Michelson's hand calculation results.

2. This behavior did not result in unacceptable consequences and the core is not calculated to uncover for the small break accident scenarios postulated by TVA (Michelson).

3. Applicants and licensees will be required to include, as part of their ongoing re-evaluation of plant response to small breaks (a) analyses of breaks which exhibit repressurization with subsequent pressurizer valve failure, and (b) documentation of analyses and data which support the conclusion that steam condensation-induced structural loadings are bounded by the large break LOCA structural loadings.

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* In the TMI-2 case, the "break" was the failed relief valve.

B+W 205 - Decay Heat
Removal after small-break

COMPUTED DATE
CHECKED DATE

1. Introduction

The Nuclear Systems Analysis Section recently undertook a qualitative examination of the decay heat removal problem associated with recovery of a Combustion Engineering System 80 pressurized water reactor (PWR) plant from a very small break loss-of-coolant accident (LOCA). A very small break LOCA is one for which the steam generators must remove a significant portion of the decay heat; otherwise, reactor coolant system repressurization occurs since the break is too small to facilitate the transport of all decay heat to the environs. In this class of LOCA's, depressurization rates are relatively slow (when compared to those normally analyzed as small breaks) and these might lead to inadequate makeup rates from the high pressure injection pumps. An ongoing qualitative consideration of this concern

QUESTION 14. Mr. Denton, the morning of March 30 you stated you had advised the state police to evacuate out to five miles. On what information did you base that recommendation for evacuation? Did you make the same recommendation to the Governor at that time?

ANSWER. (NRR-Denton)

At that time I had been advised that a helicopter had flown into the plume near the plant and had measured radioactivity levels of 1.2 rems/hour. There was some indication that additional releases could occur. Given the relatively high levels of radiation reported and concerns about the ability to control or prevent further releases, I concluded that evacuation was a prudent course of action. I so advised other members of the NRC staff and suggested that the State of Pennsylvania be notified by the NRC's Office of State Programs.

QUESTION 15. On Friday, March 30, there were releases of radioactive gases into the air. What benefits would there have been for the public in evacuating after notification that a release had occurred?

ANSWER. (NRR)

The highest radiological exposures that may have been received by some members of the public, less than 100 millirem whole body, were substantially smaller than the dose criteria recommended by the U.S. Environmental Protection Agency as warranting protective actions. These criteria, 1,000 to 5,000 millirem whole body, have been published by the EPA staff in a "Manual of Protection Guides and Protective Actions for Nuclear Incidents," EPA-520/1-75-001, 1975. Although an evacuation following notification that a release had occurred might have resulted in some lowering of exposure, it is likely that such a benefit would have been overshadowed by the risks inherent in evacuation.

QUESTION 16. Did you feel you had adequate information during the initial stages of the Three Mile Island incident to advise the Governor on an evacuation decision? If not, at what point do you feel you had adequate information to give advice on this decision?

ANSWER. (NRR-Denton)

No, I don't feel the NRC had adequate information concerning the accident during the early stages. I didn't feel comfortable with the level of information available until after I had met with my staff on Friday night at the TMI-2 site. At that time my staff had gone through the plant and was able to give me first-hand information on the status of the core, the containment, the effluent treatment system and radiation levels. Chairman Hendrie informed Governor Thornburgh on Friday morning that the Commission was also concerned about the adequacy of the information the Commission had received.

From Friday night on, I was able to obtain the benefit of expert advice about the possible course and consequences of actions needed to bring the TMI facility to a safe shutdown condition and about the consequences of further problems that might arise from such actions.

QUESTION 17. Looking back now with hindsight, who do you feel was in the best position to advise the Governor on evacuation during Wednesday and Thursday of the incident? during the following days?

ANSWER. (NRR-Denton)

The situation during Wednesday and Thursday indicates the need to improve this area. No one in retrospect appeared to be in a very good position to advise the Governor. Perhaps each licensee needs an incident center near the site which could be manned by technical staff of the licensee, representatives of the NRC, and representatives from State and local governments. After I arrived at the site and had support of a large number of NRC experts, I thought I was in the best position to advise the Governor.

One of the "lessons learned" is that an emergency evacuation is best managed from the site. Those at the site are in the best position to evaluate the situation and should be the ones advising the decision-makers.

QUESTION 18. What kind of information do you feel the Commission needs to make a recommendation on evacuation? Was the Commission getting that kind of information?

ANSWER. (NRR)

The Commission, or other authorities, involved in evacuation decisions, need to have information regarding the potential for projected radiological exposures to exceed protective action criteria together with information permitting a judgment to be made that the reduction in exposure that could be expected as a result of an evacuation would be sufficient to offset the risks inherent in evacuation. Information of this nature was sparse at the NRC Incident Response Center, and communications among the utility, the State, the NRC staff, and the Commission were garbled at times, and some recommendations were confusing and contradictory. One of the lessons learned from the accident is the need to improve the quality of information available to the various parties in the public safety decision process and the quality of its communication.

QUESTION 19. The Commission spent a considerable amount of time discussing the possibility of a precautionary evacuation. What kind of situation, do you feel, would have warranted a precautionary evacuation?

ANSWER.

A precautionary evacuation would have been warranted if there was a significant increase in the amount of fission product activity being released from the facility. For example, the NRC recommendation for the precautionary evacuation of children and pregnant women was based on the potential of a significant release from the auxiliary building. If a significant threat to the integrity of all fission product barriers had developed, a wider scale precautionary evacuation would have been warranted. The attached tables reflect some of the types of events and evacuation considerations that were developed by the NRC staff at its Incident Response Center in Bethesda during the Three Mile Island accident.

(revised from first draft - April 1, 1979) Attachment Q19

April 2, 1979

NRC PROCEDURES FOR DECISION TO RECOMMEND EVALUATION

Who Decides

1. Combination of consequences and times require immediate initiation of evacuation: Senior NRC Official on site recommends to Governor.

2. Unplanned event with substantial risk takes place or is imminent or situation judged excessively risky but there is time for consultation. Senior NRC Official notified Governor and NRC HQ. Chairman makes recommendation to Governor after consulting with Commissioners if possible.

3. Planned event involving significant additional risk. Chairman and Commissioners makes recommendation.

QUESTION 20. On April 23, 1979, Governor Thornburgh testified before this Committee that there are proven hazards in evacuating people - particularly those under medical care. Was NRC aware of these hazards and, if so, were they taken into consideration in the recommendations for precautionary evacuations?

ANSWER. (NRR-Denton)

In my recommendation regarding evacuation on Friday morning, I was considering only avoidance of radiation exposure and the injury to significantt numbers of people that might have resulted if no action were taken. At that time, I did not attempt to balance the benefits to many against the risks to a few that I knew could result from any evacuation. However, the hazards in evacuating people were considered every time the question of evacuation was discussed by the Commission. In subsequent meetings with the Governor and his staff, I came to a better appreciation of the complexities involved in planning and accomplishing evacuation, especially for those who are ill and elderly and those having difficulty with farm animals. Such factors are clearly important where evacuation may yield only marginal reductions in exposure to the balance of the affected populace.

QUESTION 21. When was the Governor first contacted by the Commission on the evacuat'on situation? Was this soon enough?

ANSWER. (SP/MPA/ELD)

On Friday, March 30, 1979 at 10:07 a.m., Chairman Hendrie (as spokesman for the Commission) first cont ed Governor Thornburgh and recommended for people, in the northeast direction . the plant to a distance of about five miles, to stay indoors. Later that morning because of uncertainties about continuing releases of radioactive material and possible bursts, Chairman Hendrie, acting on the Commission's decision, recommended to the Governor that a precautionary evacuation of pre-school children and pregnant women within 5 miles of the plant could be useful.

With respect to what actually happened, the less than satisfactory communication situation severely hampered the Commission's decision whether to recommend an evacuation. Reliable and consistent information from monitoring equipment was difficult to obtain. Readings, taken several hours before the Chairman contacted the Governor, were reported and estimated a dose of 1.2 rems per hour in the plume over the plant. Calculations indicated that by the time the plume reached the ground outside the containment, the maximum off-site dose would be 120 millirems, which 's below EPA evacuation trigger levels.

Currently, licensee and State emergency plans have no provisions for the Commissioners to contact anyone, much less the Governor of a State with recommendations on protective measures. All licensee, State and local plans call for actions between these parties and for them to notify NRC. These plans do not specify what the Commission's role should be during a nuclear accident. The Commissioners contacted the Governor as soon as they learned from staff that the Governor heard the NRC was recommending evacuation.

QUESTION 22. Was the Commission aware that Mr. Collins had advised the Civil Defense director at 9:15 a.m. March 30 to evacuate? Did the Commission concur in this recommendation? What information did Mr. Collins base his recommendation on? Was this recommendation for evacuation warranted? Who's responsibility at NRC is it to make a recommendation regarding evacuation?

ANSWER. (SP)

The Commission was not aware at about 9:15 a.m., March 30, that Mr. Collins had telephoned the Director of the Pennsylvania Emergency Management Agency, Col. Oran Henderson, with the recommendation concerning evacuation. The Chairman and the other Commissioners found out about this recommendation later on in the morning. Mr. Collins, who was the senior representative from the NRC Office of State Programs in the NRC Operations Center based his recommendation on the recommendations to evacuate which were being voiced by several NRC Senior Management personnel in the Center at the time. He inquired of them as to whether or not they wanted him to transmit this recommendation to Pennsylvania authorities, and the answer was in the affirmative. Based upon the situation which was perceived to exist at the plant site that morning, the difficulty in knowing precisely what was occurring, the recommendation, at the time, does not appear to have been unwarranted as a precautionary measure.

At the time of the TMI accident there was no specific assignment of responsibility within NRC for making recommendations on evacuation. NRC Manual Chapter 0502, "NRC Incident Response Program", does place the responsibility for major decisions affecting NRC's response actions on the NRC Emergency Management Team, consisting of the Executive Director for Operations, and the Directors of Nuclear Reactor Regulation, Inspection and Enforcement, and Nuclear Material Safety and Safeguards. The special and sensitive nature of recommendations and decisions on evacuation requires that more specific procedures be developed relative to NRC's role and where responsibilities for performing this role should be placed. This will be done in the current revision of the NRC's agency plan for dealing with emergencies.

QUESTION 23. Before the Three Mile Island event, did NRC have a set of criteria which would trigger, for example, a precautionary evacuation or an evacuation of pregnant women and children? Would such criteria have been helpful in this situation? What criteria should NRC have for determining whether to recommend an evacuation?

ANSWER. (SP)

Criteria for taking protective actions (evacuation and sheltering) in an event of this type have been recommended and published by the staff of the U.S. Environmental Protection Agency. ("Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," EPA-520/1-75-001, 1975). These criteria have been available to and utilized by the NRC since their publication and were well known to NRC staff members at the Incident Response Center during the Three Mile Island accident. However, EPA Protective Action Guides are technical guides or criteria given in terms of ranges of numerical values for projected radiological dose. At the present time, the NRC staff sees no need to change the basic technical criteria, but better planning appears necessary to assure their proper implementation.

The tables in the attachment to question 19 reflect some of the types of events and evacuation considerations that were developed by the staff at the Incident Response Center during and after the TMI accident. Precautionary evacuation and/or selective evacuation (such as pregnant women and pre-school age children) undoubtedly will be in the future, based also upon social considerations which are non-technical in nature and judgmental in character.

QUESTION 24. Mr. Denton, on Friday morning, March 30, in Bethesda you recommended a precautionary evacuation. Friday afternnoon at the Three Mile Island site you felt there was no immediate need for it. What were the main factors influencing this decision?

ANSWER. (NRR-Denton)

I had changed my views after arrival at the Three Mile Island site as a result of the understanding the staff had obtained of the source of the radioactive material being released and the means for reducing and controlling the releases and resulting offsite doses. From that point on, I believed that any decision on evacuation could and should await the development of circumstances where a release was imminent. Through the actions of the utility and the staff that circumstance did not arise.

QUESTION 25. On Saturday, March 31, you were concerned about the hydrogen bubble and what means to use to attempt to start the reactor towards cold shutdown. Did the Commission have in mind at that point any kind of threshhold level which would trigger evacuation?

ANSWER. (NRR-Denton)

By Saturday, a number of methods had been devised to remove the bubble from the primary system. On Saturday I had in mind a view that certain types of contingency measures, such as attempts to remove the bubble through depressurization and residual heat remover cooling should be attempted only after careful planning for potential evacuation. I considered that if such measures were necessary, a change in the basic cooling mode of the reactor should be made only in the daytime at an announced time and with an ability to evacuate if events proved that action to be necessary.

A "threshold level" of radiation which would trigger evacuation was not established. The staff assured the Commission during this entire time that the core cooling situation was sufficiently stable that if something unexpected were to happen it would be preceded by warning signals that would allow time for evacuation.

QUESTION 26. What kind of coordination did NRC have with the Federal Disaster Preparedness Agency during the Three Mile Island Accident?

ANSWER. (SP)

It is not clear from the question if the organization referred to is the Federal Disaster Assistance Administration or the Federal Preparedness Agency. Both are included in the answer.

In the case of the Federal Disaster Assistance Administration (FDAA), the NRC had close coordination. The Administrator of FDAA visited the NRC Operations Center in Bethesda on March 31 and set the stage for this coordination at about the same time that a memorandum from Jack Watson, Special Assistant to the President, instructed the Administrator to set up an Operations Center at the site to coordinate the Federal assistance effort. For approximately one week thereafter, the FDAA had a representative present at the site and in the NRC Operations Center (Bethesda) around the clock.

The NRC had no coordination as such with the Federal Preparedness Agency. That Agency kept abreast of the situation through daily status reports from the NRC Operations Center in Bethesda.

QUESTION 27. Did you feel the monitoring that was done throughout the Three Mile Island event provided adequate information so that the decisions made on evacuation sufficiently protected the health and safety of the public?

ANSWER. (NRR)

Field monitoring done throughout the Three Mile Island accident was an important input to the recommendations made on evacuation but was not the primary basis. The monitoring indicated that protective action criteria were not exceeded. However, on the morning of March 30, 1979, there was a measured plume of radioactivity of about 1.2 rems/hours several hundred feet above the ground. Its apparent source was the occasional discharge from the waste gas vent header in the auxiliary building that was caused by continued letdown flow from the primary system. There was a recognized potential for continuous discharge from this source within a few hours if there was continued operation of the reactor in this letdown mode, given the limited waste gas decay tank capacity.

In the earliest stages of the accident, monitoring information on the plant itself was provided to the Pennsylvania authorities, in accordance with the prepared plans, and formed the basis for the earliest decision making on the matter of evacuation which is known to us. In retrospect, this information was adequate to protect the health and safety of the public, but it is clear that better and more timely information from onsite monitoring should be available in the future and should be better coordinated with offsite monitoring as the latter becomes available in the evaluation of an accident.

QUESTION 28. Both Friday and Saturday, March 30 and 31, there were conflicting press reports as to whether the NRC had ordered an evacuation and what kind they were recommending. What factors contributed to this conflicting information?

ANSWER. (NRR)

Probably the principal factor was that the press was receiving information from a variety of sources during a time when the knowledge of the accident and its consequences were changing rapidly. These sources had varying degrees of technical information available to them, and complications caused by the interpretation of partial sets of data undoubtedly led to differing views of its significance.

QUESTION 29. How much time, given the worst possible case of a core melt, did NRC feel there would be to carry out an evacuation?

Given that amount of time, was the NRC confident that the State of Pennsylvania could carry out their evacuation plan?

ANSWER. (SP/NRR/RES)

Emergency response is warranted in the event of any radiological accident where projected doses could exceed the EPA protective action guides (PAGs). Potential accidents(e.g., core melt) which can exceed the PAGs, could begin within a range of about one half hour to many hours or days after the initiating event. It is not necessarily the intent of emergency response to prevent a radiological exposure--in most situations the intent of emergency response is to prote the public health and safety by reducing public exposure. In addition, emergency response is not limited to evacuation but includes sheltering and other protective measures to achieve its objective. NRC believes that even in the event of an accident with little or no initial warning, emergency response (sheltering and evacuation) could be effective in reducing public exposure.

With respect to the TMI accident specifically, on Friday, March 30, 1979 the NRC staff estimated there would have been at least 6 hours to carry out an evacuation. Now, staff estimates, using an accident sequence model, that at least 16 hours would be available from the time of loss of all cooling of the core (if that had occurred) until imminent failure of the containment building.

The Pennsylvania Emergency Management Agency indicated to us that its evacuation plans had been based on an area of about 5 miles in radius. This apparently was what they were prepared for on March 28, 1979. We think that they could have handled the 5-mile radius evacuation reasonably well at that time. Toward the end of the week (after March 30) we had greater confidence that Pennsylvania State evacuation plans had been adequately developed and expanded. It was not until toward the end of the week that local government authorities were adequately apprised of their role in these expanded State plans for 10 to 20 miles in radius from the Three Mile Island site.

QUESTION 30. What impact would there be on evacuation if this worst possible case were to happen at night? Would more time be needed for evacuation? If so, how much more time? What role do you feel the press played in helping or hindering the dissemination of information to the public regarding recommendations on evacuation? What improvements could be made here?

ANSWER. (SP/PA)

If an accident occurred at night, most people would probably be at home, thus making notification and warning easier. Night driving would likely be no more troublesome than the problems posed by heavier traffic during daylight hours. However, the presence of rain, fog or snow at night could make the evacuation process more difficult because of restricted visibility and snow on the highways.

Aside from weather considerations, more time would probably not be needed for night time evacuation because of the "balancing out" factors mentioned above. This assessment is not based on any extensive study or analysis of evacuation and would be more valid in cases, such as TMI, where the emergency was a developing situation and as such allowed time for anticipating a possible decision to order or recommend an evacuation.

Information on Governor Thornburgh's recommendation that pregnant women and small children leave the five mile area around the plant (no evacuation was ordered) was reported accurately by the press. This announcement was handled by the Governor's office after the Governor talked with Chairman Hendrie. Since this was an area where the Governor had the responsibility, NRC did not issue a statement on the matter. In the ensuing days, the Governor continued to be the spokesman on this matter. The only improvement we would suggest is the general observation that it would have been enormously helpful if both the Governor and the NRC had better information early in the accident. We believe the addition of direct telephone lines from the NRC operations center in Bethesda to the control rooms of operating plants and the work NRC is doing to locate off-site emergency operations centers at each plant will improve the situation considerably. But we still expect the State to take the lead in discussing matters such as evacuation or movement of people.

QUESTION 32. Does NRC consult with any Federal agency such as EPA or HEW on the evacuation question?

ANSWER. (OCM/IE)

During the morning of March 30, when NRC was considering a recommendation for evacuation, it did not discuss this matter with other Federal agencies. However, a part of the consideration by the NRC involved the protective action guidelines which are established by EPA. The Commission and the NRC staff discussed recommendations regarding evacuation with the Governor and officials of the Commonwealth of Pennsylvania, Friday, March 30, 1979.

Later that same day, Commissioners Gilinsky and Bradford met with representatives from the White House, EPA, HEW, FDA, and the National Cancer Institute at HEW headquarters to discuss evacuation capabilities and responsibilities of all affected parties, offsite monitoring procedures, data coordination, and current status of the reactor at TMI. The agencies represented offered to assist the NRC in any way they could in developing the Federal response to the accident.

A second meeting was held on March 31 at the White House to update the discussions held the previous evening.

QUESTION 33. There was considerable discussion by the NRC Commission and staff about the radius which might have to be evacuated -- 5, 10 or 20 miles. o you feel that you have sufficient data to establish the radius which must be evacuated to protect the public health and safety?

ANSWER. (SP/IE/NRR)

There were discussions of precautionary evacuations to several possible distances. These discussions had as background many years of generic work on emergency planning for power reactor facilities. In the most recent work in this area, the establishment of a generic radius describing a proposed Emergency Planning Zone for the plume exposure pathway has been recommended by an NRC/EPA Task Force in its recent report (NUREG-0396, EPA 520/1-78-016). This radius is about 10 miles for light water nuclear power plants of about 1000 Megawatts-Electric in size. Protective measures that are envisioned for this zone, if they were needed are: evacuation of part or all of the zone; sheltering in part or all of the zone; combinations of evacuation and sheltering; and/or the administration of thyroid blocking agents to members of the populace residing in the zone or a part of it.

The degree to which any one, combinations, or all of these measures would be implemented during an emergency situation would be determined by the conditions which prevailed at the time of a radiological release or conditions which might be projected for the zone. This kind of information was available during the discussions of precautionary evacuations at TMI-2, but better information and better communication is desirable for the future, as described in answers to other questions.

The "about 10 mile" radius does not imply the establishment of a radius around an area in which evacuation as a protective measure is envisioned to the outer limits of the zone in each case of an accidental radiological release. Rather, the proposed Emergency Planning Zone of about this radius, describes an area where pre-planning ought to take place for implementing all or part of the protective measures previously mentioned.

In summary then, the proposed Emergency Planning Zone is an area in which best effort is performed making use of existing emergency planning guidance concerning protective measures.

The status of the NRC/EPA Task Force Report, NUREG-0396 and its recommendations, are given in the answer tc question number 51.

QUESTION 34. There were varying estimates as to the time it would take for an evacuation. How much time do you feel would be realistic for evacuating an area of 5 miles around TMI? A radius of 20 miles?

ANSWER. (SP)

Here we must rely on the estimates of State and local authorities since evacuation times depend upon a variety of factors determined by the individual State and local situations. In this instance, the State of Pennsylvania estimated that it would take about three hours to evacuate the 26,000 people within the five mile area around TMI and about ten hours to evacuate the 700,000 people living within a 20 mile radius of the plant. In both these cases, the estimates were made taking into account that the hospitals, nursing homes and penitentiary would be essentially evacuated prior to the general movement of the population. The estimates prepared by the State of Pennsylvania are probably realistic.

QUESTION 35. What recommendations would you have for improving NRC's role in carrying out their responsibilities in advising on possible evacuations? For the State in carrying out their responsibilities? For other Federal agencies?

ANSWER. (SP/IE)

One area which requires improvement is communications. The difficulty in obtaining prompt information on the physical plant status and health physics conditions in and around the site during the TMI accident has led the NRC to the conclusion that dedicated telephone lines to each site are necessary. NRC has already had AT&T install dedicated lines to all operating nuclear power plants. Other actions now in progress that will improve NRC's capabilities in making recommendations on evacuation are the improvement of radiological monitoring around nuclear power plants and instrumentation to follow the course of an accident. Other areas for improvement or clarification of roles are being examined and recommendations will be forthcoming.

In addition, the NRC will develop internal procedures as part of its Incident Response Program which would include the chain of authority for making recommendations, the minimum data required on radiological releases, population distribution and other factors that should be in hand, the sources of such data and the means of verifying them.

The States should have their evacuation procedures for areas around nuclear facilities developed in advance. These procedures should be worked out with local government authorities. Plans should also be established for providing information to the public in the event of accidents. In particular, means for providing current information to the people directly affected by an event and its possible consequences is necessary.

As part of NRC's incident response procedures, we should determine appropriate assisting roles of other Federal agencies and include them in agreements with these agencies.

QUESTION 36. Dr. Mattson, on March 30 you recommended an immediate evacuation. On what information did you base this recommendation and do you now feel this would have been the prudent thing to do?

ANSWER. (NRR-Mattson)

My recommendation to evacuate on Friday morning, March 30, was based upon the availability of two types of information, as follows:

a) There was a plume of radioactive gases leaving the site and measured to be 1.2 rems/hr in the plume. Its source was the gaseous releases from the vent header in the waste gas system in the Unit 2 Auxiliary Building earlier that morning. The intermittent releases were expected to become a steady source of release within one to two hours because of diminishing waste gas decay tank storage capacity. A steady source of release from the vent header would have significantly increased the radiation level in the offsite plume.

b) There was a bubble of noncondensible gas (believed to be hydrogen) in the reactor coolant system. Its volume was inferred from indirect measurements to be about 1000 cubic feet at 1000 psi. It was the judgment of the technical staff that I directed in the Incident Response Center that upon depressurization of the reactor coolant system to reach the operating range of the decay heat removal system (about 300 psi), this bubble would expand sufficiently to displace water from the core and inhibit core cooling for a dangerously long period of time.

We were informed that the licensee was considering depressurization of the reactor coolant system and initiation of the decay heat removal system in order to avoid a continuous release from the vent header. Faced with either alternative a or b, above, and lacking another alternative, I recommended evacuation. I believe today that, based on the information available at the time, it was the correct recommendation. I recommended an evacuation of up to 10 miles. At the time, I was informed of the general population distribution within a 10 mile radius of the site, and I was aware of the public safety risks (e.g., traffic accidents and trauma for the ill or aged) inherent in any evacuation.

Subsequently, the radioactive releases from the vent header were brought under control so that the reactor coolant system could be maintained at high pressure without large offsite releases. This permitted the initiation of degassing operations over a period of several weeks to remove the noncondensibles in the reactor coolant system. Thus, by later in the day Friday, a general evacuation proved to be unnecessary. Subsequently, we have learned that the information available to the Incident Response Center on the potential for a steady, uncontrolled release from the vent header was apparently not valid.

QUESTION 37. Early in the Three Mile Island event the Director of the Pennsylvania Emergency Management Agency told the Governor that they had the capability to conduct evacuation. Was the NRC confident from the beginning that this plan would be effective if an evacuation had been ordered?

ANSWER. (SP)

The Pennsylvania Emergency Management Agency indicated to us that its evacuation plans had been based on an area of about 5 miles in radius. This apparently was what they were prepared for on March 28, 1979. We think that they could have handled the 5-mile radius evacuation reasonably well at that time. Toward the end of the week (after March 30) we had greater confidence that Pennsylvania State evacuation plans had been adequately developed and expanded. It was not until toward the end of the week that local government authorities were adequately apprised of their role in these expanded State plans for 10 to 20 miles in radius from the Three Mile Island site.

QUESTION 33. Could you briefly describe your regional and resident inspector programs for operating nuclear power plants and those under construction? To what extent do your inspectors actually observe a licensee's operations and activities?

ANSWER. (IE)

Inspections are performed on power reactors under construction, in test and in operation. Prior to the implementation of the Resident Inspection Program, inspections were conducted exclusively from the five regional offices by two categories of NRC inspectors, generalists and specialists. The generalists, sometimes called principal inspectors, have overall responsibility for more than one plant and they also assist in inspecting other plants. Generalist inspectors often possess specific technical expertise. These generalists have been the group from which the Resident Inspectors have been chosen. Specialists are experts in specific technical disciplines, such as health physics, physical security or heavy construction techniques, and they conduct inspections in these specialized areas in support of the more general inspection.

Inspections are part of NRC's review of applications for licenses as well as NRC's issuance of construction permits and operating licenses. Inspections continue throughout the operating life of a nuclear facility.

Prior to construction, the inspection program concentrates on the applicant's establishment and implementation of a quality-assurance program. Inspections cover quality-assurance activities related to design, procurement and the plans for fabrication and construction. An acceptable inspection finding is a prerequisite for NRC's acceptance of an application by a potential licensee. After an application has been accepted for review, inspections continue and acceptable inspection findings are an important part of the NRC's decision to issue a construction permit.

During construction, a sampling of licensee activities is inspected to make sure that the requirements of the construction permit are followed and that the plant is built according to design and applicable codes and standards. Construction inspections look for qualified personnel, quality material, conformance to approved design and for a well-formulated and satisfactorily implemented quality-assurance program, since these factors are most important to the successful construction of a nuclear plant. The licensee's implementation of these elements is assessed by examination, on a spot check basis of construction activities.

As construction nears completion, preoperational testing to demonstrate the operational readiness of the plant and its staff begins. Inspections during this phase determine whether the licensee has developed adequate test plans, assure that tests are consistent with NRC requirements and determine that the plant and its staff are prepared for safe operation. Inspections during the preoperational phase involve (1) reviewing overall test management procedures; (2) examining selected test procedures for technical adequacy; and

(3) witnessing and reviewing selected tests to determine their outcomes and the consistency of planned and actual tests. In addition, inspectors review the qualifications of operating personnel and assure that operating procedures and quality-assurance plans are developed and implemented.

About six months before the operating license is issued, a startup phase begins in preparation for fuel loading and power ascension. Following the issuance of an operating license, fuel is loaded into the reactor and the actual startup test program begins. As in preoperational testing, NRC inspection emphasis is placed on test management procedures and results. The licensee's management system for startup testing is examined, test procedures are analyzed, tests are witnessed and licensee evaluations of test results are reviewed. Inspectors also independently evaluate licensee activities.

When startup testing is completed satisfactorily, routine operations begin. Thereafter, NRC continues its inspection program throughout the operating life of the plant to verify that the licensee's control systems assure the safe operation of the plant in compliance with NRC requirements. Specific elements of the operating reactor inspection program are:

Review of the basic systems and procedures the licensee follows to be certain they conform with requirements and are technically sound and implemented properly.

Analysis of records of licensee operation and interviews of personnel to confirm that actions called for by the prescribed systems and procedures are routinely followed.

Periodic verification of licensee and system performance by means of independent NRC observation, tests or measurements.

In addition to the inspection of nuclear power reactor licensees, the Dallas Regional Office also conducts a program of audit inspection of contractors and vendors who provide services and components to the nuclear industry. This program, the Licensee Contractor and Vendor Inspection Program (LCVIP), is directed toward assuring that services and products of selected licensee contractors and vendors are controlled by good quality-assurance programs. This assurance is achieved through direct NRC field inspection and investigation of these programs. Corrective action is taken when necessary.

Regular inspections of these nuclear steam system suppliers, architect engineers, fuel fabricators and major manufacturers of components are conducted. Firms are selected for inspection based on the significance of the product to plant safety and the manufacturer's volume of active nuclear business. Inspectors examine hardware, review documents and interview personnel to ensure that detailed control procedures have been prepared and are being followed. These inspections do not relieve licensees of any responsibility for accepting individual components but, as mentioned above, are intended to verify that contractors and vendors have good quality-assurance systems. In addition to the regular or preventive inspections, investigations are conducted under the LCVIP as a result of allegations, inspection reports or licensee reports the NRC receives. Data obtained about problems experienced in operating plants is fed back into the LCVIP to minimize repetitions and correct generic problems which may be undetected in other plants.

In June 1977, the Commission approved a revised inspection program that includes stationing NRC inspectors onsite at nuclear power reactor sites having units in operation, startup, or preoperational testing and at selected sites having units in the later stages of construction. In addition to resident inspectors, the revised program includes:

A national performance appraisal capability that provides three elements: (1) evaluation of the performance of NRC licensees from a national perspective; (2) an evaluation of the effectiveness of the NRC inspection program; and (3) confirmation of the objectivity of NRC inspectors.

A significant extension of direct verification of licensee activities by NRC inspectors that involves more direct measurement and increased observation of operations and tests in progress.

An enhanced career management program.

The resident inspectors will be similar to the generalists (or principal inspectors) in the current program and they will conduct general inspections over a broad area ranging from activities of the reactor operators to the health physics and physical security programs. However, in-depth technical inspections will be conducted by specialists who will continue to be assigned in the regional offices.

Initial information from the first group of resident inspectors at reactor sites indicates that the amount of time they spend at the plant has increased by a factor of about two. This increased time should provide greater opportunity to observe and measure licensee activities, verify licensee compliance, and respond to significant events. Furthermore, the resident inspectors have shown improved knowledge of the details of the plant assigned. Consequently, inspectors should be able to provide better technical judgments concerning that plant and improve the effectiveness of inspections.

It is difficult to accurately quantify the amount of time inspectors actually observe a licensee's operations and activities. An inspector is trained to observe a licensee's operations and activities during his inspections, i.e., checking in at the site, walking through the plant, talking with plant staff, etc. In addition, 20 percent of an inspector's onsite time (about 30% of his total available time) is planned as independent inspection effort in areas where the inspector has particular expertise or concern.

QUESTION 39. What is the basis for NRC's inspection approach of verifying that the licensee has a system in place for meeting all NRC requirements rather than independently verifying that those requirements are met?

ANSWER. (IE)

Under the total NRC regulatory program, the licensee is clearly responsible for the safety of his plant. The thrust of the NRC inspection program is to emphasize this responsibility to the licensee and to assure that he carries it out. We believe that the licensee can best discharge this responsibility systematically.

In examining the licensee's discharge of this responsibility, the NRC does independently verify the effectiveness of implementation of the licensee's systems by audit-type inspection utilizing independent verification. We have emphasized the importance of independent verification by NRC inspection in our inspection program for some time now and intend to increase this emphasis in a programmed fashion.

Audit inspection is a time proven method of inspection that leaves the auditor in an objective position. In operating from the position of an objective auditor, the inspector does not leave the licensee in a position where he perceives that his responsibilities have been diluted. From the position of objective auditor, the inspector can aim his efforts at independent verification of whether or not the system under consideration is effective and what might be necessary to correct it.

QUESTION 40. When it is available, would you please provide the following information.

QUESTION a. Do you know whether the valves to the auxiliary feedwater pumps for Unit 2 were closed during NRC's inspections in March?

ANSWER. (IE)

An inspection was conducted by the Office of Inspection and Enforcement on March 19 to 23 and 26 which covered activities at Unit 1 and Unit 2. The activity inspected at Unit 2 was limited to review of information concerning licensee event reports that were forwarded to the Commission; no control room or plant tour was conducted. Therefore, the inspector did not physically observe either the position of the valves in the plant or the position indicator for the valves in the control room. However, on March 23, an NRR Operator Licensing Branch Examiner was conducting operator examinations of the control room at TMI 2, and did in fact select walkdown of the emergency feedwater valve alignments in the control room as part of his oral examination. He was thereby able to verify in this manner that the valves were ope at that time.

QUESTION b. Were the switches in the control room tagged at the time?

ANSWER. (IE)

Our review of the situation revealed that a "tag" was not applied to the switches for the valves in question. A caution tag was applied to a controller not associated with the auxiliary feedwater valves which is located just above the switches and indicating lights for the subject valves (EFV-12B and EFV-12A). The caution tag for the controller is of such a size that it apparently could have obscured both the switch and indicating lights for valve EFV-12B, only.

QUESTION c. Wouldn't it have been a simple matter for either the plant operator or an NRC inspector to check those tags to ensure that the switches were in the correct position?

ANSWER. (IE)

It would have been a simple matter to lift the tag which was applied with a string to observe the valve switch and indicating lights. An NRC inspector was not in the control room of Unit 2 at the time that the tag was applied and he did not enter the control room until after the incident on March 28, 1979.

QUESTION 31. Do you feel the people of Pennsylvania had enough information to make their own informed decision on whether or not to leave the Three Mile Island area?

ANSWER. (NRR-Denton)

I do not feel that the people of Pennsylvania had sufficient information to make their own informed decisions regarding evacuation during the first few days since the condition of the core and the amount of radioactivity that had been released to the containment and auxiliary building had not been well characterized for the public at that time. I believe one way such situations might be improved would be to devise some way of making available to the public uniform and objective analyses and data about the accident. The early publication of general assurances of no danger or general warnings of imminent catastrophe do not provide an adequate substitute for such factual information about the accident and the implications of planned actions at the plant. The daily Preliminary Notices issued by the NRC regarding the accident provided a useful vehicle for conveying this type of information.

QUESTION 41. Does the reactor operator have the responsibility at the beginning of his shift to check the instruments and controls in the plant to verify that the plant is operating in accordance with the requirements of his license?

ANSWER. (IE)

See attached excerpt from TMI-2 Administrative Procedure, AP 1012, "Shift Relief and Log Entries," (Paragraph 3.7, "Shift Relief").

There is no regulatory requirement for an NRC licensed operator to check the instruments and controls in the plant at the beginning of his shift. He is responsible as a matter of practice, to be aware of plant status. This is included, normally, in a company administrative procedure, such as that attached.

QUESTION. Are there any indications from your investigation thus far that the operators at Three Mile Island did not meet this requirement?

ANSWER. (IE)

The investigation thus far indicates that the operators at Three Mile Island acted in accordance with the "Shift Relief" requirements specified in AP 1012, paragraph 3.7.

3.7 Shift Relief

3.7.1 All shift operations personnel shall be responsible for maintaining their duty station until properly relieved. The Shift Supervisor, Shift Foreman, Control Room Operators and Auxiliary Operators shall be relieved by qualified personnel only, e.g. those personnel who are properly licensed and properly informed of the plant status, operations in progress, and any special instructions which may be applicable. The relieving individual will discuss the plant status, operations in progress and special instructions with on-duty personnel so that he is adequately informed prior to assuming his shift duties.

3.7.2 The Control Room Operator will acknowledge his understanding and awareness of the changes in the plant status since his own last entry by signing the Control Room Log prior to assuming the shift duty.

3.7.3 During his shift the relieving individual shall insure adquate review of station logs, records, special instructions, etc., which have been generated since his last shift. The logs and records to be reviewed should include:

1. Shift Foreman Log
2. Control Room Log
3. Hourly Computer Log
4. Tagging Application Book
5. Equipment and Fuel Status Boards
6. TCN and SOP Books
7. Standing Order Book
8. Operations Memo Book
9. Preventative Maintenance Schedule Books
10. Revision Review Book

QUESTION 42. Do you believe there is value in increasing the amount of time which is available to regional inspectors to independently observe the licensee's activities? Would doubling that amount of time be a feasible and useful measure?

ANSWER. (IE)

We believe that increasing the amount of time which is available to achieve a higher level of independent verification aimed directly at operations and operational readiness of safety related equipment is of value. We have spent a number of years structuring, modifying and testing our present program which includes some independent verification of licensee activities. We believe that the best way to achieve a higher level of independent verification is by adding a resident inspector at each unit in the operation, start-up, or preoperation testing phase, whose primary duty would be to inspect, observe, verify and witness licensee activities. Expanding resident coverage is much more efficient than conducting the same program from an expanded region based operation.

We believe the additional amount of time that could be provided by placing a resident at each unit would give us more than a doubling of the current independent verification effort and that it would be a feasible and useful addition.

QUESTION 43 I understand that NRC's present schedule for its resident inspector program calls for a resident inspector at each operating and construction site by fiscal year 1982. Doesn't the Three Mile Island accident indicate that this schedule should be accelerated? What are the difficulties in accelerating this program? Would nine months from now be a feasible date for having a resident inspector at each operating unit, with a resident inspector at each unit under construction by the end of fiscal year 1980? Why not?

ANSWER. (IE)

The NRC is currently expanding the resident inspection program at nuclear power reactor sites. (See answer to question #44). With this expansion a total of 145 resident inspectors will be assigned to all sites having one or more reactors in preoperational testing, startup, or actual operation. This will amount to 70 sites by the end of FY 1981. Another 25 resident inspectors will be located at reactor construction sites, mostly those where reactors are in the later stages of construction.

We would like to implement the program faster and have recently examined the possibility of doing so. Unfortunately, the rather lengthy recruitment and training lead time, the limited availability of qualified personnel coupled with the need to maintain an effective region-based inspection program, preclude implementing the program any earlier.

In our judgment, the current implementation plan is proceeding as rapidly as possible. Resident inspectors for the first two years of implementation (FY 1978 and FY 1979) have, for the most part, come from personnel already on-board before the resident program was approved and funded. The FY 1978 supplemental resources (approved for the resident program) have allowed us to recruit and train additional personnel who will be assigned to sites in FY 1980 and FY 1981. Consequently, it is not feasible to either have a resident inspector at each operating unit within nine months or at each unit under construction by the end of fiscal year 1980.

QUESTION 44: As I understand it, your plan for resident inspectors calls for only one resident inspector at multiple unit sites. Shouldn't there be one resident inspector for each unit?

ANSWER. (IE)

At the time this question was raised, the resident inspector program for reactors called for:

1 resident inspector at each reactor site having one or more units in operation, startup, or preoperational test.

1 resident inspector at selected sites with one or more units in the late stages of construction.

The NRC has subsequently expanded the resident inspection program within resources approved by the Congress, which will, by the end of FY 1981, assign

- as many resident inspectors per reactor site as there are units at that site undergoing preoperational testing or startup and/or actually operating, with a minimum of 2 residents for each such site;

- resident inspectors to selected sites having one or more units in various stages of construction;

Thus, there will be as many operations resident inspectors as there are units at a site (excluding units under construction).

QUESTION 45(a): Could you describe for us NRC's ongoing investigation of the TMI nuclear accident.

ANSWER. (MPA)

The NRC investigation incorporates three complementary components. The first, conducted by the Office of Inspection and Enforcement (IE) was limited to two aspects of the accident: (1) those related to the actions taken by the licensee before and during the accident, and (2) steps taken by the licensee to control radioactive releases off-site, and steps to implement the licensee emergency plan. The results of the IE investigation were published in August, 1979 as NUREG-0600: "Investigation into the March 28, 1979 Three Mile Island Accident by Office of Inspection and Enforcement" (see Attachment 1).

The second component of the NRC investigation, conducted by the Office of Nuclear Reactor Regulation, had a different purpose: "To identify and evaluate those safety concerns... that require licensing actions... for presently operating reactors as well as for pending... applications." The technical scope of the report covered:

1. Reactor operations, including operator training and licensing;

2. Licensee technical qualifications;

3. Reactor transient and accident analysis;

4. Licensing requirements for safety and process equipment, instrumentation and controls;

5. Onsite emergency preparations and procedures;

6. NRR accident response role, capability and management; and

7. Feedback, evaluation, and utilization of reactor operating experience.

The results of the NRR investigation were published in July, 1979 as NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short Term Recommendations" (see Attachment 2).

The third element is the overall Commission investigation known as the "NRC/TMI Special Inquiry Group." The group is examining the history of the NRC review of Metropolitan Edison's application to operate TMI-2; the inspection history; the emergency response by NRC including staff and Commissioners; the coordination among Federal, State and local officials, the utility, industry sources, and national labs; and the implications of the accident for the licensed nuclear power program .

QUESTION 45(b): Who will be conducting the investigation?

ANSWER. (MPA)

The Commission has employed Mr. Mitchell Rogovin, of Rogovin, Stern and Huge, to direct the Special Inquiry. The Deputy Executive Director for Operations, Dr. Kevin Cornell, is the senior NRC member of the Inquiry. The directors of the Offices of Inspection and Enforcement and Nuclear Reactor Regulation were in charge of conducting their own investigations.

QUESTION 45(c): Will there be a separate task force within NRC to conduct the investigation so as to permit the NRC and its staff to continue their other responsibilities?

ANSWER. (MPA)

The overall Commission investigation has a complement of 65 NRC and 10 contractor professional support personnel working full time. Creating the task force has not hampered the work of the NRC offices substantially.

IE dedicated 15 investigatores to its investigation, and NRR dedicated about 70 professionals to its investigation.

QUESTION 45(d): What impact has the Three Mile Island accident had on NRC's ability to continue to meet its other responsibilities in a timely and effective manner? Will there be any effect on NRC's licensing reviews and the schedules for those reviews for power plant construction permits or operating licenses?

ANSWER. (MPA)

The accident has had a severe adverse impact on the ability of NRR to perform its scheduled work. The staff announced a moritorium on licensing. A full assessment of the extent of the impact on schedules is not yet complete.

NUREG-0600

INVESTIGATION INTO THE MARCH 28, 1979 THREE MILE ISLAND ACCIDENT BY OFFICE OF INSPECTION AND ENFORCEMENT

Investigative Report No. 50-320/79-10

Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission

NUREG-0573

TMI-2 LESSONS LEARNED TASK FORCE STATUS REPORT AND SHORT-TERM RECOMMENDATIONS

Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission

QUESTION 46. What will be the budgetary impacts of the Three Mile Island accident for NRC, both for fiscal year 1979 and for fiscal year 1980. Will a supplemental authorization and appropriation be necessary? When will this information be available?

ANSWER. (CON)

For FY 1979, the reprogramming approved by your Committee on August 2, 1979 will provide sufficient funding for the remainder of this year. For FY 1980, the Commission is proposing a supplemental appropriation for TMI-related activities. This supplemental request has been provided to OMB for their review and approval. We have requested OMB to take early action on this supplemental so that it may be transmitted to Congress before adjournment.

Question 47

A witness at our last hearing on the Three Mile Island accident testified that it is conceivable as one reads the transcripts of the Commission meetings, that public relations was playing an unwarranted role in the decision-making process. That witness referred specifically to an exchange between Commissioners Gilinsky and Ahearne. Chairman Hendrie, I would appreciate your comments as well as those of Commissioners Gilinsky and Ahearne.

Answer

The question refers to Mr. Roisman's prepared statement, page 11 (enclosed) in which it is alleged that the Commission during a closed meeting on April 1, 1979 urged withholding from the public "a report prepared by one of NRC's leading safety experts on the maximum consequences which might occur as a result of explosion of the large hydrogen bubble at the plant." The document was not a report on the potential consequence of the explosion of hydrogen but was rather a first draft of a decision document on whether and what type of evacuation should be recommended in a variety of situations. A copy of the first draft is enclosed. The transcript (which was prepared from a tape recording) shows that the document referred to by Commissioner Gilinsky was one drafted by a staff group at the direction of Commissioner Gilinsky and with the participation of some of the Commissioners and brought to the meeting by Dr. Hanauer (enclosed pages 20, 21, 22 of the transcript). The Commissioners had not finished redrafting the document. The second and third draft plans (copy enclosed), which were developed during and after the meeting include significant revisions to the first draft.

Commissioner Ahearne states that at the time of the referenced comments by Commissioner Gilinsky and himself, the Hanauer document was a draft that had not had two important reviews:

First, it had not been reviewed by the senior NRC technical people actually at Three Mile Island. The draft had been prepared in Bethesda and had not been reviewed by Harold Denton or Victor Stello, who were much more familiar with the technical aspects of the problem than were the Bethesda staff. Nor had it been examined by the other senior technical person in the Commission, Chairman Hendrie. Thus, Commissioner Ahearne was concerned the draft could very likely contain significant technical errors.

Second, the document outlined how the Governor would be notified in an emergency. Since the document had not been reviewed by Harold Denton, these steps had not been discussed with the Governor to get his comments. In addition, Mr. Denton had developed a good working relationship with the Governor, which could be very important in achieving rapid response in the case of an emergency. Therefore, Commissioner Ahearne believed the Governor should receive the draft from Mr. Denton, not from the press.

To summarize, Commissioner Ahearne believed the draft should be checked by Denton and Hendrie and be given to the Governor before it was released to the press.

Enclosures:
1. Roisman testimony
2. First April 1 draft of NRC Decision Procedures
3. Transcript of April Commission meeting
4. Second April 1 draft of NRC Decision Procedures
5. April 2 draft of NRC Decision Procedures

Testimony of

ENCLOSURE 1

Anthony Z. Roisman
Staff Afforney
Natural Resources Defense Council

Before the

Subcommittee on Nuclear Regulation
of the
Senate Public Works Committee

I. Introduction

My name is Anthony Z. Roisman and I am a staff attorney with the Natural Resources Defense Council. For many years NRDC has been actively involved in providing through litigation, administrative action and legislation an increased public awareness of the risks associated with the use of nuclear technology to generate electricity and of the advantages of less risky alternatives to nuclear power. The events at the Three Mile Island nuclear facility provide all of us with graphic evidence not only of some of those nuclear risks but also of how incompletely we and the experts charged with designing, building, operating and regulating nuclear power plants understand those risks. Because even now the accident at Three Mile Island continues, because substantial dangers still remain and because the accident, its causes and consequen are not yet fully understood, many urge that it is too earl to draw any conclusions from the events at Three Mile Island. I believe there is already sufficient evidence available to learn some imporatant lessons from this continuing catastrophe, and my testimony today will focus on those lessons:

...ittee on the Biological Effects of Ionizing Radiation, Dr. ...ward Radford, as well as Drs. Arthur Tamplin and Thomas Cochran of the NRDC staff, believe are at least ten times too high). These certain health effects to workers should have been included in the government assessments.

It is reprehensible that even while the Three Mile Island accident continues the apparently irresistible urge to downplay the health and safety consequences of this accident and the risks of nuclear power persists. In a revealing exchange at one of the Commission meetings on Three Mile Island two Commissioners urged witholding from the public a report prepared by one of the NRC's leading safety experts on the maximum consequences which might occur as the result of explosion of the large hydrogen bubble at the plant (from unedited transcript of closed Committee meeting, April 1, 1979, p. 21):

"COMMISSIONER GILINSKY: YOU JUST KEEP IT TO YOURSELVES. IT SHOULD NOT GO OUT OF HERE."

"COMMISSIONER AHEARNE: IT SHOULD NOT GO TO THE PRESS, FOR EXAMPLE."

It is too late to put a happy face on nuclear technology and it is high time that NRC and the nuclear industry begin telling it like it is.

III. Conclusion

The aftermath of Three Mile Island will include a series of autopsies of the accident each designed to find and reveal the truth about what happened, why and what to do about it. In

April 1, 1979

This table includes a number of assumptions about activity and weather. These assumptions have been chosen conservatively. In an actual release, the release rate and weather should be evaluated as they are at the time, and the decision based on those values.

Decision Sequence

Event - Spontaneous failure or decision to perform a potentially risky maneuver.	In tables
Find out what actually happened and what is functioning (1 hour)	
Predict what could result - different likelihoods	
Predict release rate	
Determine present weather and forecast	Assumed constant in table
Dose prediction	In table
Action Guidelines	Per Appendix 7

Who Decides

NRC decision is made by Mr. Denton, who is also Presidential representative. There are two parallel paths of information and analysis leading to Mr. Denton or his representative:

Path 1: NRC man in control room - Open line to NRC Incident R sponse Center. NRC technical people at Center phone communication to Denton.

Path 2: Another NRC man in Control Room. Open line to NRC trailer, NRC technical people in trailer. Denton in same trailer or in communication with it.

(April 1 forecast)

	EVENT	EXPECTED PLANT RESPONSE (RANGE?)	RELEASE AND TIME	WARNING TIME	EVACUATION SCENARIO	WHO DECIDES
1.	Loss of vital function or decision to perform a potentially risky maneuver. Examples 1. Reactor Coolant Pump Trip. 2. Leak in Aux Building. 3. Loss of off-site power. 4. Loss of feedwater. 5. Depressurization to go on RHR.	Restore Function within 1 hour	No significant change		~~None~~ Evac, No shelter	NRC recommends to State Governor
		Switch to Alternate Function involving Pri Coolant in Aux Building	Small leak in aux building less than 1 gal/hour		~~None~~ Evac Shelter 2 mi (1)	
			Large leak in aux building 50 gal/min	2 hour	Evac 2 miles (2) Stay Inside 5 miles	
		Serious possibility of failure to restore a a vital function	Core melt; see item 2 below & Appendix			

*For sufficiently risky maneuver, do precautionary evac 2 mi and stay inside 5 mi; whether to do this or not depends on details of maneuver and plant situation. There is also the potential for a precautionary evacuation on more general grounds.

	EVENT	EXPECTED PLANT RESPONSE (RANGE?)	RELEASE AND TIME	WARNING TIME	EVACUATION SCENARIO	WHO DECIDES
2.	Core Melt	Maintain Containment* Integrity (likely) with Containment Cooling	Tech Spec Containment Leak Rate	4 hour	Precautionary Evac 2 mi all around and 5 mi sector,** stay inside 10 mi (3)	NRC recommends to State Governor
		Containment headed for Breach	Reactor Safety Study Categories PWR 4 - See Appendix	24 hour	Evac 5 mi all around & 10 mi sector, stay inside 15 mi (4)	
3.	Hydrogen Explosion Inside Reactor Vessel	Mixture in explosive range			Precautionary 2 mi (?)	
		No significant change in reactor or primary system	No significant change		None	Info from Control Room to NRC Representative via paths analysis by 2 NRC groups in parallel
		Core Crushed (unlikely)	Core melt See item 2 & Appendix			
4.	Evacuate Control Room (except very temporarily)	Loss of Control	Probably caused by core melt		Evac 5 mi all around and 10 mi sector** stay inside 15 miles (4)	

**90° Sector

Action Alternatives

	Evacuation	Stay Inside
1.		2 miles
2.	2 miles	5 miles
3.	2 miles all around 5 miles 90° sector	10 miles
4.	5 miles all around 10 miles 90° sector	15 miles

a. All sector choices governed by wind direction. If shifting, more than one quadrant may be affected.

b. These are initial values; as the release continues measurements may indicate the need for reconsideration of action up to 20 miles.

Action Guidelines

a. Notify evacuation authorities two hours in advance (if available) to standby for a possible evacuation.

b. Projected ~~Predicted~~ doses of 1R rem whole body or 5R rem thyroid, ~~in 2 hours~~ - stay inside.

c. Projected ~~Predicted~~ doses of 5R rems whole body or 25R rems thyroid, ~~in 2 hours~~ - mandatory evacuation of all persons.

Assumes general warning already that some form of evacuation may become necessary.

Weather

The table is based on a conservative prediction of the weather for the next few days, based on the April 1 forecast. At the approach to decision time for evacuation, the appropriate materological condition will be factored into the dose estimates to determine the evacuation time, sectors, and distances for the evacuation.

NRC is predicting the dispersion characteristics of the region for the currently measured meteorology as the incident progresses.

Heat Generation

The reactor core is now quite cool compared to the conventional design-basis calculations.

1. The reactor is new, so no fuel has more than 3 months equivalent operation, compared to 1-2 years average for other plants.

2. The neutron chain reaction has been shut down for over 4 days.

It should also be noted that the concrete basemat of this plant is unusually thick.

As a result of the above differences, calculations for this plant at this time predict that the core will not melt its way through the containment.

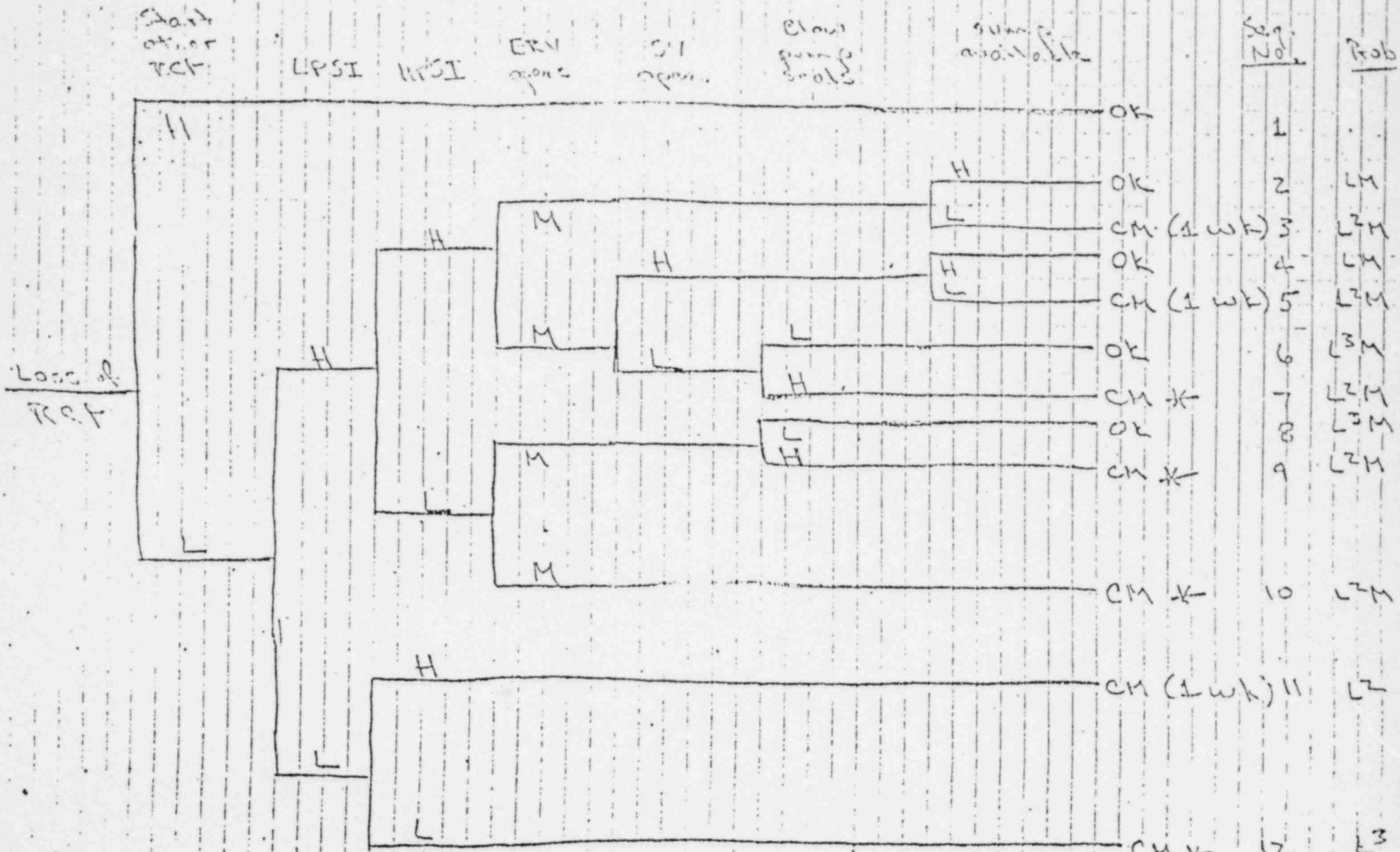
Major sequences evaluated here are tied to the loss of forced circulation in the RCS. The loss of flow from the reactor coolant pump (RCP) is the generalized initiating event from which other initiating events such as loss of offsite power can develop.

APPENDIX 1.a SEQUENCES OF POSSIBLE SYSTEMS FAILURES

Figure 1.b-1 shows the loss of RCP event tree. This tree shows the various options available given the loss of the RCP, and indicates which combinations of events or failures would lead to core meltdown (CM). The sequences denoted with an asterisk are those which would be expected to follow the core meltdown progression discussed below, leading to the variety of atmospheric radioactive releases and consequences discussed later. Some core meltdowns could be expected to be delayed for roughly a week because of the availability of ECC injection over that period. This method of core cooling, however, is not expected to be adequate to prevent core melt; as such a core meltdown is assessed to occur at roughly a week. A rough measure of relative probabilities of the various outcomes is indicated by the notation of L, M, H (low, medium, high). The column on the right-hand side of the page indicates the relative probabilities of the sequences, with "LM" as the highest probability and L^3M as the lowest.

Fig. 1.b-1.

Loss of Reactor Coolant Pump



MAJOR EVENTS AND TIMING IN EVENT OF CORE MELTDOWN

Event 1 - Sprays and Coolers Operative

Time=0 Flow stops, core and water start heat-up

Time=100 min Core starts to uncover

Time=150 min Core begins to melt

Time=200 min Molten core is in lower head of reactor vessel, pressure is 2500 psia

Time=210 min Reactor vessel fails, containment pressure goes to 25 psia

Time=210 min Hydrogen burns, containment pressure goes to 67 psia
Steam explosion possibility - minor consequence

<u>CONTAINMENT SURVIVES</u> (Failure assumed 130 psia)

Time=10 hours Molten core has melted about 1 meter into basemat

Time=days Major problem - handle hydrogen, oxygen - maintain containment integrity

CAUTION:
- Keep sprays running
- Keep water many feet over molten debris
- WITHOUT RECOMBINERS Hydrogen continues to build up

<u>BASEMAT SURVIVES</u>

Event 1 Conclusion: This event should not produce major releases

Event 2 - Sprays and Coolers Failed Before Flow Stops

Time=0 to Time=210 min. Same as Event 1 - containment pressure is 25 psia

Time=810 min Containment pressure is 70 psia

Time=1 day Containment fails due to steam (mostly) overpressure - about 135 psia

<u>CONTAINMENT FAILS</u>

Event 2 Conclusion: This event leads to major releases.

The event tree for core melt leading to various releases is shown in Figure 1.b.

The following are essential in the event of core melt.

1. Sprays and coolers are required to prevent major releases.
2. Hydrogen must be recombined or otherwise removed from containment.

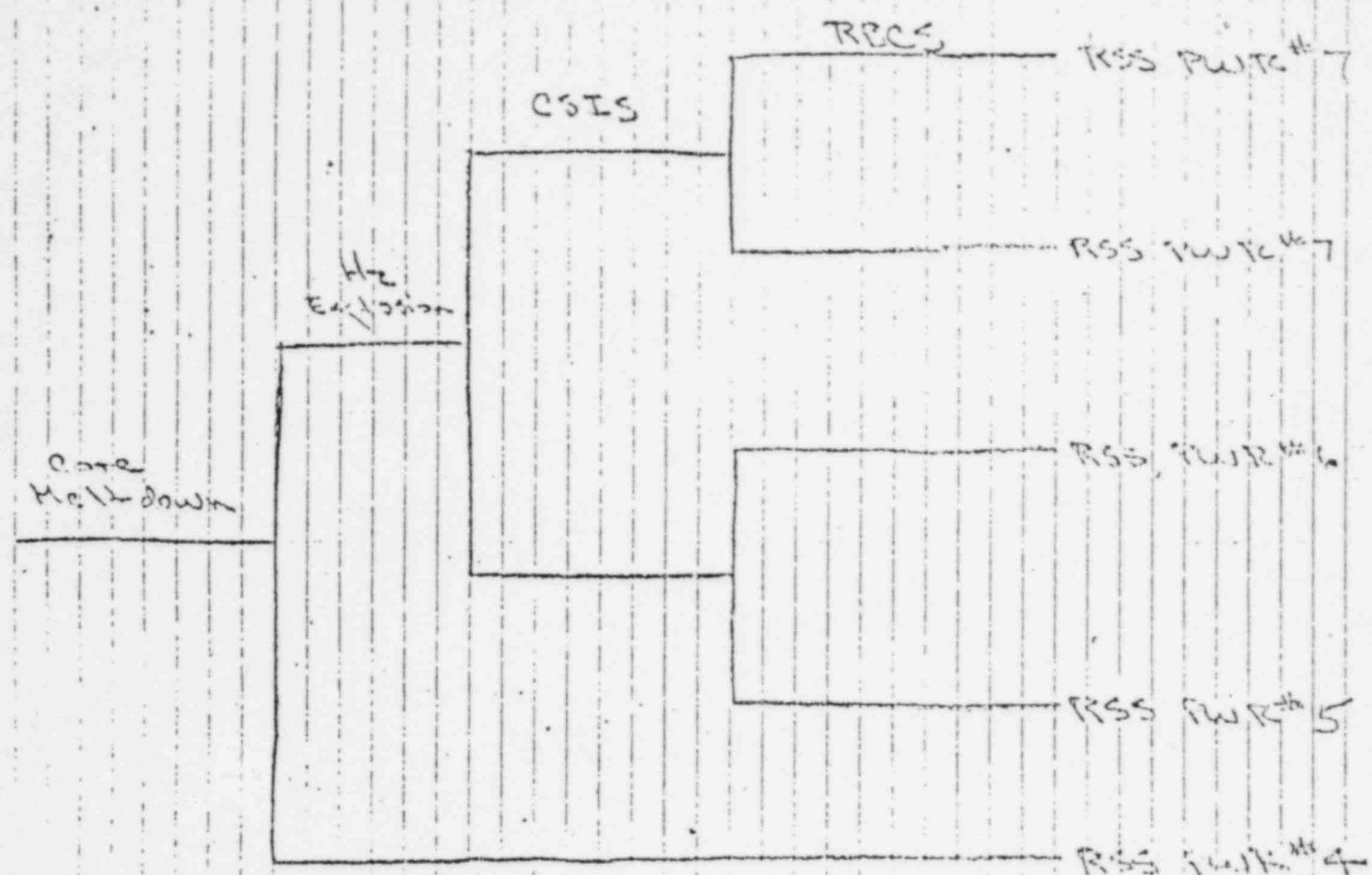

Core
Meltdown
H2
Explosion
CSIS
RECS
RSS PWR #7
RSS PWR #7
RSS PWR #6
RSS PWR #5
RSS PWR #4

FIGURE 1.b

c. Large Leak in Auxiliary Building (AB)

The activity level in the reactor coolant is so high that substantial releases can come from small amounts spilled in the AB which requires once through ventilation. A leak of 5 gpm to the AB atmosphere is assumed for the expected level of leakage. A leak of 50 gpm is taken as a large leak to consider a major leak in pump shaft sealing or some similar mishap. Based on the leakage experienced already only the noble gases and no iodine are assumed to evolve. The AB ventilation exhaust is assumed to flow through the charcoal filters.

d. Hydrogen Explosion in Reactor Pressure Vessel

A detonation of the hydrogen oxygen bubble in the reactor vessel could rupture the vessel and/or crush the core. Rough analysis indicates that the pressure vessel would not rupture. Postulation of the core response is difficult. If the core is crushed, it could effectively prevent core cooling leading directly to the core melt sequence described earlier. It is unlikely that compression would lead to criticality.

ENCLOSURE 3

NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF:

CLOSED MEETING

Place -

Date - Sunday, 1 April 1979 Pages 1 - ~~69~~ 75

(THIS TRANSCRIPT WAS PREPARED FROM A TAPE RECORDING.)

Telephone:
(202) 347-3700

ACE-FEDERAL REPORTERS, INC.
Official Reporters
444 North Capitol Street

yesterday's worst case is high in calculation --

DR. HANAUER: The worst?

COMMISSIONER AHEARNE -- was 14,000 psi.

DR. HANAUER: But that on a vessel like that, the impulse, the energy in that is pretty small, in fact. It's a very high spike and it'll probably stress, but the strength to those short pulses is very high.

COMMISSIONER AHEARNE: Joe had reached the conclusi[illegible] it would cause rupture.

DR. HANAUER: It would be easier to bend - well, it not my area.

COMMISSIONER GILINSKY: (Inaudible) --

COMMISSIONER AHEARNE: Pardon me?

COMMISSIONER GILINSKY: -- keep saying if it went -

COMMISSIONER AHEARNE: No, hey, I had asked him specifically and he had concluded that would rupture. That's a worst case calculation on the spike, sort of an effect of detonation.

DR. HANAUER: Even it were larger --

COMMISSIONER GILINSKY: (Inaudible)

COMMISSIONER KENNEDY: What level, John?

COMMISSIONER AHEARNE: 14,000 psi.

DR. HANAUER: If, even if it ruptures, it takes a lot of energy to make some pieces fly through that thick containment.

COMMISSIONER AHEARNE: Right.

COMMISSIONER KENNEDY: Energy that, that level you don't think is available?

DR. HANAUER: I haven't done the calculation. What I've seen leads me to think well I don't see pieces flying and breaking containment.

COMMISSIONER KENNEDY: Joe concluded that, too. He did not think that there would be a free missile that would breach containment.

COMMISSIONER BRADFORD: He had had an impression of the core you're talking about - that it did heat up (inaudi it would restart a metal water reaction.

COMMISSIONER KENNEDY: Yes.

COMMISSIONER BRADFORD: And how much more hydrogen would that make available?

DR HANAUER: (to another person) You've got all I have now. You can't have that. That one's mine. No I gave them to the gentleman behind me.

(Simultaneous discussion.)

COMMISSIONER GILINSKY: You just keep it to yourse It should not go out of here.

COMMISSIONER AHEARNE: It should not go to the pre for example.

MR. KENNEKE: I only need two.

DR. HANAUER: I have no use for more than one. Al the masters are right out here in the office if you need more

COMMISSIONER BRADFORD: I'd asked about how much mo hydrogen should we get -- could get if we restarted.

DR. HANAUER: I don't know the numbers. You have - I've seen a lot of these calculations and there are a lot of assumptions in them and if the core melts and there isn't muc water around it, you've got all your metal-water, if it melts more slowly and goes plop, plop, plop you could get a lot.

I've seen calculations with like 30 percent metal-water, and we don't have anything like that much. I really can't give you--I just don't know. There's a rule of thumb somebody was using this morning that we've had about 4 percer and that would say you could get maybe five, six, eight time as much as you have now.

But I'm really on very shaky ground. I hate to gi people numbers that felt good, you know.

COMMISSIONER GILINSKY: What about this pregnant women business?

DR. HANAUER: Are you going with me or do want to stay?

COMMISSIONER GILINSKY: No, no -- this question to

COMMISSIONER BRADFORD: This question is do you believe that if you evacuated out this one ring of the popul as whole, would you go further for the especially susceptibl

DR. HANAUER: Well again, I'm only brokering opini

NRC PROCEDURES FOR DECISION TO RECOMMEND EVACUATION

Who Decides

1. Combination of consequences and times require immediate initiation of evacuation: Senior NRC Official on site recommends to Governor.

2. Unplanned event with substantial risk takes place or is imminent or situation judged excessively risky but there is time for consultation. Senior NRC Official notifies Governor and NRC HQ. Chairman makes recommendation to Governor after consulting with Commissioners if possible.

3. Planned event involving significant additional risk. Chairman and Commissioners makes recommendation.

Unplanned Events

	EVENT	EXPECTED PLANT RESPONSE	RELEASE AND TIME	WARNING TIME	EVACUATION SCENARIO
1.	Loss of vital function or unplanned leaks.	Restore function within 1 hour	No significant change		Possible precautionary evac 2 mi; stay inside 5 mi
	Examples Reactor Coolant Pump Trip;	Switch to Alternate Function involving Primary Coolant in Auxiliary Building	Small leak less than 1 gal/hour		possible precautionary evac 2 mi; stay inside 5 mi
	Loss of offsite power;		Large leak 50 gal/min	2 hour	Evac 2 miles Stay Inside 5 miles
	Loss of feedwater; Depressurization to go on RHR; Leak in Auxiliary Building	Serious possibility of failure to restore a vital function See 2			

— conservative

These tables include a number of assumptions about activity and weather, chosen realistically. In an actual release, the release rate and weather should be evaluated as they are at the time, and the decision based on those values.

	EVENT	EXPECTED PLANT RESPONSE	RELEASE AND TIME	WARNING TIME	EVACUATION SCENARIO
2.	Sequence leading to Core Melt	Maintain Containment Integrity (likely) with Containment Cooling	Design Containment Leak Rate	4 hour	Precautionary Evac 2 mi all around and 5 mi, 90° sector, stay inside 10 mi
		Containment expected to Breach	Significant release of core fission products	24 hour (time for containment failure)	Evac 5 mi all around and 10 mile, 90° sector, stay inside 15 mi
3.	Hydrogen flame or explosion possible inside reactor vessel	Mixture in flammable range			Precautionary 2 mi (?) + 5 mi 90° sector 10 mi stay inside
		Explosion; major damage Core Melt See 2			
4.	Evacuate or Lose Control Room	Loss of Control Treat like major release			Precautionary (?) 2 mi Evac 5 mi all around and 10 mi 90° sector, stay inside 15 miles

EVENT	EXPECTED PLANT RESPONSE	RELEASE AND TIME	WARNING TIME	EVACUATION SCENARIO
Planned Manuever.	Probability of losing vital function	See releases under loss of vital function	Timing of maneuver can be set to provide as much time as necessary	Precautionary evacuation 2 miles, stay inside 5 miles PLUS See outcomes under loss of vital function.

Action Guidelines

a. Notify evacuation authorities two hours in advance (if possible) to standby for a possible evacuation.

b. Projected doses of 1 rem whole body or 5 rems thyroid stay inside.

c. Projected doses of 5 rems whole body or 25 rems thyroid mandatory evacuation of all persons.

Assumes general warning already that some form of evacuation may become necessary.

Weather

The table is based on a realistic prediction of the weather for the next few days, based on the April 1 forecast which would result in high doses at a given distance. At the approach to decision time for evacuation, the appropriate meterological condition will be factored into the dose estimates to determine the evacuation time, sectors, and distances for the evacuation.

NRC is predicting the dispersion characteristics of the region for the currently measured meteorology as the incident progresses. Rain could lead to higher local radioactivity levels.

Heat Generation

The reactor core is now quite cool compared to the conventional design-basis calculations.

1. The reactor is new, so no fuel has more than 3 months equivalent operation, compared to 1-2 years average for other plants.

2. The neutron chain reaction has been shut down for over 4 days.

It should also be noted that the concrete basemat of this plant is unusually thick.

As a result of the above differences, calculations for this plant at this time predict that the core will not melt its way through the containment.

MAJOR EVENTS AND TIMING IN EVENT OF CORE MELTDOWN

Event 1 - Sprays and Coolers Operative

Time=0 Flow stops, core and water start heat-up

Time=100 min Core starts to uncover

Time=150 min Core begins to melt

Time=200 min Molten core is in lower head of reactor vessel, pressure is 2500 psia

Time=210 min Reactor vessel fails, containment pressure goes to 25 psia

Time=210 min Hydrogen burns, containment pressure goes to 67 psia
Steam explosion possibility - minor consequence

<u>CONTAINMENT SURVIVES</u> (Failure assumed 130 psia)

Time=10 hours Molten core has melted about 1 meter into basemat

Time=days Major problem - handle hydrogen, oxygen - maintain containment integrity

CAUTION:
- Keep sprays running
- Keep water many feet over molten debris
- WITHOUT RECOMBINERS Hydrogen continues to build up

<u>BASEMAT SURVIVES</u>

Event 1 Conclusion: This event should not produce major releases

Event 2 - Sprays and Coolers Failed Before Flow Stops

Time=0 to Time=210 min Same as Event 1 - containment pressure is 25 psia

Time=810 min Containment pressure is 70 psia

Time=1 day Containment fails due to steam (mostly) overpressure - about 135 psia

<u>CONTAINMENT FAILS</u>

Event 2 Conclusion: This event leads to major releases.

ENCLOSURE 5

NRC PROCEDURES FOR DECISION TO RECOMMEND EVALUATION

Who Decides

1. Combination of consequences and times require immediate initiation of evacuation: Senior NRC Official on site recommends to Governor.

2. Unplanned event with substantial risk takes place or is imminent or situation judged excessively risky but there is time for consultation. Senior NRC Official notified Governor and NRC HQ. Chairman makes recommendation to Governor after consulting with Commissioners if possible.

3. Planned event involving significant additional risk. Chairman and Commissioners makes recommendation.

UNPLANNED EVENTS

EVENT	EXPECTED PLANT RESPONSE	RELEASE AND TIME	WARNING TIME	EVACUATION SCENARIO
1. Loss of vital function or planned leaks.	Restore function within 1 hour	No significant change		Possible precautionary evac 2 mi; stay inside 5 mi
Examples Reactor Coolant Pump Trip; Loss of off-site power;	Switch to Alternate Function involving Primary Coolant in Auxiliary Building	Small leak less than 1 gal/hour		Possible precautionary evac 2 mi; stay inside 5 miles
Loss of feed-water;		Large leak 50 gal/min	2 hours	Evac 2 miles Stay Inside 5 miles
Depressurization to go on RHR; Leak in Auxiliary Building	Serious possibility of failure to restore a vital function See 2			

These tables include a number of assumptions about activity and weather, which are somewhat pessimistic. In an actual release, the release rate and weather should be evaluated as they are at the time, and the decision based on those values.

EVENT	EXPECTED PLANT RESPONSE	RELEASE AND TIME	WARNING TIME	EVACUATION SCENARIO
2. Sequence leading to Core Melt	Maintain Containment Integrity (likely) with Containment Cooling	Design Containment Leak Rate	4 hours	Precautionary Evac 2 mi all around and 5 mi, 90° sector, stay inside 10 mi
	Containment expected to Breach	Significant release of core fission products	24 hours (time for containment failure)	Evac 5 mi all around and 10 mile, 90° sector, stay inside 15 mi
3. Hydrogen flame or explosion possible inside reactor vessel	Serious flammability problem			Precautionary 2 mi plus 5 mi 90° sector, 10 mi stay inside
	Explosion; major damage Core Melt, See 2			
4. Control Room Evacuation	Possible Loss of Control			Precautionary 2 mi
Plant Evacuation	Treat like major release			If plant evacuated Evac 5 mi all around and 10 mi 90° sector, stay inside 15 miles
5. Release during cleanup				

PLANNED EVENTS

EVENT	EXPECTED PLANT RESPONSE	RELEASE AND TIME	WARNING TIME	EVACUATION SCENARIO
Planned Manuever that involves a significant risk	Probability of losing vital function	See releases under loss of vital function	Timing of maneuver can be set to provide as much time as necessary	Precautionary evacuation 2 miles, stay inside 5 miles PLUS See outcomes under loss of vital function

Action Guidelines

a. Notify evacuation authorities two hours in advance (if possible) to standby for a possible evacuation.

b. Projected doses of 1 rem whole body or 5 rems thyroid stay inside.

c. Projected doses of 5 rems whole body or 25 rems thyroid mandatory evacuation of all persons.

Assumes general warning already that some form of evacuation may become necessary.

Weather

The table is based on a realistic prediction of the weather for the next few days, based on the April 1 forecast which would result in high doses at a given distance. At the approach to decision time for evacuation, the appropriate meterological condition will be factored into the dose estimates to determine the evacuation time, sectors, and distances for the evacuation.

NRC is predicting the dispersion characteristics of the region for the currently measured meteorology as the incident progresses. Rain could lead to higher local radioactivity levels.

Heat Generation

The reactor core is now quite cool compared to the conventional design-basis calculations.

1. The reactor is new, so no fuel has more than 3 months equivalent operation, compared to 12 years average for other plants.

2. The neutron chain reaction has been shut down for over 4 days.

It should also be noted that the concrete basemat of this plant is unusually thick.

As a result of the above differences, calculations for this plant at this time product that the core will not melt its way through the containment.

Event 1	Sprays and Coolers Operative
Time=0	Flow stops, core and water start heat-up
Time=100 min	Core starts to uncover
Time=150 min	Core begins to melt
Time=200 min	Molten core is in lower head of reactor vessel, pressure is 2500 psia
Time=210 min	Reactor vessel fails, containment pressure goes to 25 psia
Time=210 min	Hydrogen burns, containment pressure goes to 67 psia -- Steam explosion possibility -- minor consequences

<u>CONTAINMENT SURVIVES</u> (Failure assumed 130 psia)

Time=10 hrs	Molten core has melted about 1 meter into basemat
Time=days	Major problem -- handle hydrogen, oxygen -- maintain containment integrity

CAUTION:
- -- Keep sprays running
- -- Keep water many feet over molten debris
- -- WITHOUT RECOMBINERS Hydrogen continues to build up

<u>BASEMAT SURVIVES</u>

Event 1 Conclusion: This event should not produce major releases

Event 2 -- Sprays and Coolers Failed Before Flow Stops

Time=0 to Time=210 min	Same as Event 1 -- containment pressure is 25 psia
Time=810 min	Containment pressure is 70 psia
Time=1 day	Containment fails due to steam (mostly) overpressure -- about 135 psia

<u>CONTAINMENT FAILS</u>

Event 2 Conclusion: This event leads to major releases.

QUESTION 48. In the wake of the Three Mile Island accident, the Advisory Committee on Reactor Safeguards has renewed its earlier recommendations to the Commission that a high priority be given to research to improve reactor safety. What improved safety systems research initiatives beyond the program approved by the Commission last year are warranted in light of the Three Mile Island accident?

ANSWER. (RES)

Current evaluation of the Three Mile Island accident has resulted in the identification of the need to increase priority of some of the projects recommended by the Commission to improve reactor safety, as well as identifying other research initiatives which should be pursued.

The impact of TMI on NRC's overall research program will involve additional studies of both PWR and BWR anticipated transients and small LOCA's, plant response to accident conditions, post-accident examination of safety system components and fuel, improved instrumentation for accidents, fission product release and transport, primary coolant and containment chemistry during accidents, hydrogen behavior, plant data bank and risk assessment studies.

The previously recommended research projects related to improved in-plant accident response and vented-containment should be given increased attention in light of Three Mile Island. Expanded research effort on monitoring and diagnostic systems to assist the operator under accident conditions, improved operating and emergency procedures for responding to accident situations, and improved use of simulators in studying operator response to accident situations and for related training is clearly warranted.

QUESTION 49. Were there any factors which inhibited the effectiveness of the operation of NRC's emergency response center in Bethesda during the Three Mile Island accident? If so, what can be done to improve the ability of NRC headquarters to respond to such emergency situations?

ANSWER. (IE)

We believe that the Headquarters emergency organization and the Operations Center worked reasonably well. However, planning for the operation of the Center had not adequately considered the large size of the staff or the extent of communications needed. Expansion into nearby offices provided space. However, by doing so, the capability of recording all telephone calls was lost. Also, during the earlier stages of the response, additional attention should have been given to logging the actions of various technical support groups. As the event progressed, steps were taken to expand the telephone communications systems, arrange for the additional people at the Operations Center, implement a method of keeping track of the actions taking place and handle the large quantity of information being transmitted to the Operations Center. If a basic deficiency existed in the Operations Center, it was that planning was not adequate for an operation of the size and duration of the Three Mile Island incident.

The NRC is now looking into various ways of expanding the capabilities and resources available at the Operations Center. Several options dealing with ways to expand the type and amount of data directly available from facilities will be explored. The staff is also considering modifications to the physical layout of the Operations Center and procedures for operating the Center in order to improve its efficiency and effectiveness.

Two immediate steps have been taken with regard to the Operations Center. The first is to have the Center manned 24-hours a day to improve our emergency reaction time. In addition, direct and dedicated telephone lines have been installed in all operating nuclear power plants. Extensions of these lines are located in the Control Room, reactor supervisor's office and other locations. These lines automatically ring at the NRC Operations Center when the receivers are lifted off the telephone cradles.

QUESTION 50. What are the Commission's requirements for emergency planning on the part of a licensee, and State and local officials, to respond to an accident at a nuclear power plant? At the time of the accident, did the emergency plan for the Three Mile Island plant meet all the NRC requirements? If not, in what respects was it different?

ANSWER. (SP/IE/ELD)

The Commission's requirements for emergency planning are set forth in Section 50.34 of 10 CFR Part 50. This section requires that an applicant for an operating license provide plans for coping with emergencies, including the items specified in Appendix E to 10 CFR Part 50, "Emergency Plans for Production and Utilization Facilities." These require notification procedures and agreements with Federal, State, and local agencies in the area where each plant is located. Such agreements take congnizance of the fact that it is the State and local government jurisdictions that have the authority to implement protective action plans such as evacuation of the general public.

The NRC staff's position on 10 CFR 50, Appendix E, is set forth in Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants." In addition, NRC's siting regulations require that a low population zone (LPZ) be established for each site and require that the applicant show that protective measures could be taken within the LPZ in the event of a serious accident.

At the time of the accident, the approved licensee emergency plan of record for the Three Mile Island plant and the plant site met the requirements of the Commission's regulations.

The NRC has no regulatory requirements for emergency planning by State and local officials. We do have an active program of assisting State and local governments in radiological emergency response planning. Seven other Federal agencies cooperate with the NRC in this program. Agency responsibilities are outlined in a Federal Register Notice of December 24, 1975. The program includes issuance of guidance, training of State and local officials, field assistance to help develop and test radiological emergency response plans, and evaluation and concurrence in State and local plans. Pennsylvania did not actively seek NRC concurrence in its State plan, and it did not meet the NRC guidelines and therefore had not received NRC concurrence at the time of the Three Mile Island accident.

QUESTION 51. A recent report by the General Accounting Office states that the Commission has so far found only ten State emergency response plans for radiological releases from nuclear facilities that have all the essential elements for planning and preparedness. The report goes on to question the effectiveness in an emergency situation of untested State emergency plans and to criticize the size (5 miles) of present emergency planning zones. What are your views on these criticisms in the GAO report?

ANSWER. (SP)

NRC has concurred in 12 State plans. At the time the GAO report was prepared, NRC had concurred in 10 State plans. Presuming that any GAO criticism is based on the fact that this number is only about ¼ of the total number of States which should ultimately have these plans, and about 1/3 of the total number of States which should have these plans now, we can only say that:

(1) in spite of a lot of Federal guidance which has been in the field over the past few years, State and local governments have not put priority on this kind of planning, and;

(2) NRC and other Federal agency resources assigned to this effort have been miniscule in terms of people, funds, and other resources.

In recent months, the Chairman has written to the governors of States with nuclear power plants in operation or under construction and with no NRC concurred-in emergency plan, offering Federal assistance and urging that their plans be developed to become eligible for NRC concurrence. The responses, generally, have been favorable and the NRC field assistance effort has been significantly increased. The Chairman has also written to the heads of Departments and agencies that assist NRC in helping the States asking for an increased effort.

In addition, the matter of State and local emergency plans has been given special impetus through the report to the Commission by a Task Force on Emergency Planning and an Advance Notice of Proposed Rulemaking (44 FR 41483, July 17, 1979). The latter could result in making NRC concurred-in plans a condition for nuclear power plants to operate.

With respect to the size of emergency planning zones, a report prepared by an NRC/EPA Task Force, titled " Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants (NUREG 0396, EPA 520/1-78-016)," was published in December 1978. This report recommends the establishment of two emergency planning zones around each nuclear power plant: an inner zone of about 10 miles for the plume exposure pathway and an outer zone of about 50 miles for the ingestion exposure pathway.

Recommendations on this report are before the Commission for consideration and approval of a policy statement on guidance to State and local governments. EPA recently endorsed the guidance and will publish its policy statement in the Federal Register about September 15, 1979.

The question of the value of untested State emergency plans was addressed in the response to question 54.

QUESTION 52. If there are deficiencies in the emergency plan for a particular plant, how does the Commission justify its determination at the time of licensing that the public health and safety will be adequately protected during the operation of the plant? What was the significance of the deficiencies, if any, in the Three Mile Island emergency plan at the times that Units 1 and 2 were licensed to operate?

Answer. (NRR)

The NRC has established criteria for acceptable licensee emergency plans for nuclear power plants (see response to Question 50). The staff review of the Three Mile Island Unit 1 emergency plans was carried out predominantly in 1972 and reported in the staff's Safety Evaluation Report dated July 11, 1973. The criteria used by the staff were those found in Appendix E to 10 CFR Part 50, supplemented by a guidance document entitled, "Guide to the Preparation of Emergency Plans for Production and Utilization Facilities," dated December 1970. Revised emergency plans for the Three Mile Island site were submitted with the Unit 2 Final Safety Analysis Report beginning in May 1974. The staff review of these plans was completed in August 1975 and reported in the staff's Safety Evaluation Report dated September 1976. This review was conducted at the same time the initial Standard Review Plan (Ch. 13.3) was under development. The Standard Review Plan was published in November 1975. The criteria in effect for this review were, therefore, nearly equivalent to those subsequently published in the Standard Review Plan.

Also published for comment in November 1975 was the initial version of Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants." The criteria found in Annex A of this guide are substantially equivalent to those found in Appendix A of the Standard Review Plan. Assimilation of public comment on the Regulatory Guide resulted in publication of Revision 1 in

November 1977, and the subsequent updating of the Standard Review Plan in Revision 1 to reference the Regulatory Guide. Revision 1 of the Regulatory Guide has been used in all FSAR reviews initiated after November 1977 but, by NRC management decision (see Attachment 1), was not used to reopen reviews that had already been completed unless the applicant or licensee submitted proposed revised emergency plans.

Metropolitan Edison Company submitted such a proposed revised plan to the NRC dated May 11, 1978. As shown in the copy of the internal staff memorandum (see Attachment 2), this revised plan was found to be deficient with respect to the criteria of Regulatory Guide 1.101, Rev. 1. Attachment 2 resulted from staff review of Metropolitan Edison Company's submittal in Amendment No. 65 to their Final Safety Analysis Report for Three Mile Island Unit 2. The information request was never sent to the licensee due to internal administrative delay resulting from split responsibility within NRR for the TMI Unit 1 operating unit (Division of Operating Reactors) and the TMI Unit 2 under operating license review (Division of Project Management). Since the licensee already had an approved emergency plan, there was no perceived reason to expedite the administrative processing of the request for further revision of the TMI plan in view of other higher priority work. The delay extended up to the time of the accident.

If serious deficiencies in the emergency plan for a particular plant were known to the NRC at the time of licensing, a license would not be issued. In this respect, each prospective licensee's state of emergency preparedness is inspected by NRC's Office of Inspection and Enforcement during the months immediately preceeding the date of expected issuance of a license. On occasion, some deficiencies in preparations have been identified and subsequently corrected by the applicant before license issuance or full power operation. Following license issuance, emergency preparedness inspections are conducted annually.

As a result of the Three Mile Island accident, a special NRC staff Task Force on Emergency Planning was created. The report by this Task Force was submitted to the Commission on August 21, 1979 in a staff paper, SECY-79-499 (see Attachment 3). The recommendations of the Task Force include the development of coordinated action plans for each major NRC office. It is anticipated that implementation of the Task Force recommendations will result in an improved state of emergency preparedness at all plants.

Attachment 1

JKR

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

JAN 3 . 1977

MEMORANDUM FOR: R. Boyd, Director, Division of Project Management
R. Heineman, Director, Division of Systems Safety
V. Stello, Director, Division of Operating Reactors
H. Denton, Director, Division of Site Safety and Environmental Analysis

FROM: Ben C. Rusche, Director, Office of Nuclear Reactor Regulation

SUBJECT: REVISED PROCEDURE FOR DOCUMENTATION OF DEVIATIONS FROM THE STANDARD REVIEW PLAN

NRR Office Letter No. 2, issued on August 12, 1975, directed the staff to use the Standard Review Plan to assure consistent evaluation of all applications. It also directed that, except for clarification and correction of errors, the Standard Review Plan would remain fixed until any proposed change of substance was considered by the Division Directors, reviewed by the Regulatory Requirements Review Committee, and then authorized by the Director, NRR.

NRR Office Letter No. 9, issued on June 18, 1976, addressed the special problem associated with implementation of Office Letter No. 2 in operating license reviews when the construction permit reviews were not conducted on the basis of the Standard Review Plan guidelines. It noted the necessity to document decisions made on bases other than those defined in the Standard Review Plan and, of equal importance, the reasons for the acceptability of such bases. It then directed the staff to develop, for my approval, procedures for documenting the bases for deviations from the Standard Review Plan in each operating license Safety Evaluation, and to implement those procedures for all operating license Safety Evaluation Reports issued after January 1, 1977. My memorandum of September 20, 1976, approved an implementing procedure recommended to me by the NRR Division Directors. This procedure addressed both operating license and construction permit applications.

The experience gained in attempting to use the implementing procedure for operating license reviews nearing completion has shown that, contrary to our expectation at the time the procedure was developed, the staff is unable at this time to conform to the requirements of the implementing procedure without incurring a substantial delay in

ATTACHMENT 3

August 21, 1979

SECY-79-499

COMMISSIONER ACTION

For: The Commissioners

From: Lee V. Gossick
Executive Director for Operations

Subject: REPORT OF TASK FORCE ON EMERGENCY PLANNING

Purpose: To obtain Commission action on the recommendations of the Task Force on Emergency Planning.

Discussion: The Task Force on Emergency Planning was established in June 1979 to identify weaknesses in NRC's emergency preparedness process and to outline an approach for improving NRC's overall emergency preparedness activities. The Task Force Report, submitted on August 9, 1979, is provided as Enclosure 5. The report is being placed in the Public Document Room and will be published shortly as a NUREG document.

To assist the Commission in its review of the Task Force Report, MPA has summarized the issues, problems, and tasks described in the report. This summary is provided as Enclosure 1.

As one of its major efforts, the Task Force developed a list of 14 emergency planning issues (Enclosure 2). Public comment on these issues was solicited in a July 17, 1979 advance notice of proposed rulemaking. Comments will be analyzed and incorporated into a draft rule that will follow the usual rulemaking process. The final rule is expected to be published January 15, 1980.

Contact:
E. Hayden, MPA
49-27721

QUESTION 53. Based upon your present understanding of the situation, how well did State and local emergency preparedness officials deal with the situation at the Three Mile Island plant?

ANSWER. (SP)

The Pennsylvania Emergency Management organization was able to get a handle on the problems they faced in connection with possible offsite consequences and evacuation of the public. They needed help, which was supplied mainly by the Defense Civil Preparedness Agency in dealing with local civil defense officials. The Pennsylvania Bureau of Radiation Protection was less prepared for the situation and needed significant outside assistance (mainly from the Federal government) to get organized and carry out its primary responsibility of monitoring the environmental effects of the emergency and advising the governor in this area.

In summary the State did respond to the event within the impact areas as they perceived it. The Bureau of Radiological Health response might have been enhanced if their plans had been more developed and closely integrated with those of other agencies.

QUESTION 54. What improvements, if any, do you think are needed in the Commission's emergency planning and preparedness requirements? Would mandatory test drills of all or part of a State's emergency plan be useful and warranted?

ANSWER. (NRR/SP)

As noted in response to Question 52, the Commission's requirements for emergency planning and preparedness have been reexamined in light of the Three MIle Island experience (see SECY-79-499). This reexamination included consideration of the need to incorporate more specific requirements into the regulations, particularly with respect to such matters as (a) accident assessment with onsite instrumentation, (b) communications requirements, (c) offsite monitoring, and (d) coordination with a more explicitly defined Federal (including NRC) response role. The latter is needed to produce clarification in protective action decision making responsibilities.

The NRC's current requirements for State and local emergency planning and preparedness are agreements which licensees must make with State and local officials concerning early warning of the public and public evacuation and other protective measures. NRC, along with seven other Federal agencies does provide assistance to State and local governments in radiological emergency response planning and preparedness. This is done, in part, by issuance of planning guidance. This guidance needs to be completed where it has not been issued, and revised where it has been issued. The NRC staff intends to require offsite plans as an element of the licensing decision process. Test exercises of such emergency plans are useful and warranted to assure that all emergency response personnel have a familiarity with their roles. It is also clear that steps are needed to better inform the general public of the sources of authoritative information and instructions in the event of an emergency. One of the conditions of NRC initial concurrence in a State plan under the present voluntary program is the conduct of an acceptable test of the plan. After concurrence, an annual test of the plan is necessary to continue NRC concurrence.

QUESTION 55. Do you believe that the Federal Emergency Management Agency should assume the responsibility for making policy and coordinating radiological emergency response planning around nuclear facilities as is recommended in the GAO report?

ANSWER. (SP)

The NRC believes that FEMA should have an active policy and coordinating role in this area. However, because FEMA is newly established and has not yet had an opportunity to develop inhouse expertise in radiological emergency response planning, it would be premature for it to assume the lead role now. At least while FEMA is gaining that expertise, it will be necessary for the agencies already involved, such as NRC, EPA, DOE, and HEW, to continue providing assistance to State and local governments in emergency planning and preparedness. In this regard, the NRC is prepared to retain the functions essential to its role as nuclear regulator (e.g., for on-site monitoring and overseeing radiological training) for the interim and to re-evaluate our role when FEMA is fully organized and staffed. We welcome the establishment of FEMA and look forward to working with that agency in coordinating Federal, State and local planning and preparedness to improve protection of the public in the event of a radiological emergency.

QUESTION 55. What can be done to encourage those States with operating nuclear plants to develop a satisfactory emergency plan - those States which have not as yet done so?

ANSWER. (SP)

Letters to the governors of States with operating nuclear power plants have been signed by the Chairman. These letters give a brief status of the State radiological emergency response plan and offer the assistance of NRC in moving the plan toward early NRC concurrence. In addition, the agency with lead responsibility for this planning activity in each State has been contacted by the NRC State Programs staff. Most of these agencies have shown an interest, if not eagerness, to move ahead with development or refinement of their plan.

Temoorary assignments have been made to augment the Office of State Program staff through FY 1979. Eight permanent positions were requested in the FY 1980 supplemental. Other steps taken to improve emergency planning are included in the response to question 51.

QUESTION 57. Has there been increased interest by the States in emergency planning since the accident?

ANSWER. (SP)

Yes, a significant increase. Many States with operating reactors or with nuclear facilities near their borders, but without an NRC concurrence in their plans, have requested field assistance, plan evaluation and training. Additionally, NRC has formally offered assistance to all of these States which should have these plans.

QUESTION 58. The Advisory Committee on Reactor Safeguards wrote to the Commission on April 9, 1975 to express a number of concerns regarding emergency planning. Concerning State response plans, the Advisory Committee noted that: "the response plans of many States responsible for dealing with population groups in the neighborhood of nuclear power plants are only in the planning stages or, if completed, show a need for more professional knowledge in this subject area. Compounding these problems is the fact that Federal funds to lend support to the development of State response plans, which the Committee understood were to be made available through the Federal Disaster Assistance Administration, have never materialized."

- Are these concerns still valid today?
- What was done by the Commission to address these concerns?
- What needs to be done now to solve this problem?

ANSWER. (SP)

Many of these same concerns are valid today but to somewhat lesser degree because progress has been made in this area since 1975. Much still remains to be done.

The Commission did not do a great deal to address these concerns, primarily because this area was not previously considered one of high priority.

We need to step up Federal assistance to State and local governments, including the provision of qualified technical personnel and funds where needed for emergency plan development and to acquire preparedness resources. Priority should, and will, be given to those jurisdictions where there are operating reactors.

QUESTION 59. If there are inadequacies in response planning at the State level, what is the situation with regard to response planning for a nuclear emergency at the local government level? What needs to be done to improve the situation at the local government level?

ANSWER. (SP)

Generally, the situation at the local level is worse than at the State level. Neither the States or the NRC have placed enough emphasis on the adequacy of local government plans, and local governments do not usually have personnel or financial resources or the special expertise needed to do this type of planning.

To improve the situation the following needs to be done:

1. Generate more interest by the States in the affected local jurisdictions' planning activities.

2. Require a close tie between the State and local plans in the NRC plan concurrence process.

3. Provide funds to local governments for preparation and maintenance of their radiological emergency response plans.

4. Dedicate the necessary NRC resources to handle the substantially increased workload that would result (now about 150 affected counties and growing to in excess of 400 counties and hundreds of municipalities by 2000).