ACRS - 1721

Meeting Date: 3/5/80
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Revised Draft Issued:
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CERTIFIED

MINUTES OF THE ACRS AD HOC SUBCOMMITTEE
MEETING ON THE NRC ACTION PLANS DEVELOPED AS
A RESULT OF THE TMI-2 ACCIDENT - MARCH 5, 1980

The ACRS Ad Hoc Subcommittee on the NRC Action Plans Developed as a Result of the TMI-2 Accident, met in Room 1167 at 1717 H St. N.W., Washington, D.C. at 8:30 A.M. on March 5, 1980 to review the Near-term Operating License (NTOL) items identified in the plan and determine if they form a necessary and sufficient basis for the resumption of the issuance of operating licenses. Secondarily, the Subcommittee was to be made aware of the extent and significance of recently observed ultrasonic indications observed in the nozzles of the Sequoyah reactor vessel and to be informed of the recent transient events at Crystal River Unit 3.

Notice of this meeting was published in the Federal Register on February 19, 1980 (Attachment A). A copy of the detailed schedule of presentation is attached (Attachment B). No written statements from the public were received nor were there any requests to make oral statements (written comments from the Atomic Industrial Forum, Inc. (AIF) were submitted to the NRC and representatives from the AIF made an oral presentation to the ACRS on March 6, 1980.) No written reports were issued or approved by the Subcommittee at this meeting. A list of documents provided to the Subcommittee during this meeting is attached (Attachment C.)

Attendees:

ACRS

H. Etherington, Chairman
J. Ebersole
W. Kerr
D. Moeller
H. Lewis
W. Lipinski, Consultant
J. McKinley, Staff (DFE)

NRC STAFF

R. Mattson
R. M. Gamble
Walter Pike
Paul Vineyard
W. T. Russell
J. L. Milhoan
W. Minners

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POOR QUALITY PAGES

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ACRS

J. Stampelos, Fellow
R. Savio

NRC STAFF

R. Purple

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S. Kowkabany, Burns & Roe Inc.
V. L. Conrad, Public Ser. of Okla.
Jeffrey L. Smith, Long Island Light Co.
Betty Schellhardt, Dames & Moore
Terry C. Price, TVA
R. J. McDermott, Bechtel
T. C. Nichols, Jr. S. Carolina Elect. & Gas
Bill Williams, South Carolina
Public Ser. Authority
R. S. Boyd, KMC, Inc.
E. Zebroski, EPRI-NSAC

Executive Session (Open to the Public)

Mr. Etherington, Subcommittee Chairman, opened the meeting at 8:30 A.M. with a statement regarding the conduct of the meeting in accordance with the provisions of the Federal Advisory Committee Act and the Government in the Sunshine Act. Mr. J. C. McKinley was the designated federal employee.

Mr. Etherington referred the Subcommittee members and consultants to an internal memo dated March 4, 1980 prepared by John Stampelos, ACRS Fellow, to Dr. M. S. Plesset on his "Reviews of the NRC TMI Action Plan (NUREG-0660; Draft 2; January 23, 1980) and Examples of Possible Rush to Judgement Items" for a cross reference between ACRS concerns and the Action Plan items. It was Mr. Etherington's opinion that the action plan adequately addresses the ACRS' concerns.

Mr. Etherington pointed out that the scope of this meeting was limited to the near-term operating license items as requested in Chairman Ahearne's letter of February 19, 1980 and that the balance of the action plan would be taken up at subsequent meeting. He noted that representatives from the Atomic Industrial Forum (AIF) would make a presentation before the full ACRS on the Action Plan on March 6, 1980.

Mr. Ebersole inquired if the startup tests being required of the Sequoyah plant would include a demonstration of the "feed and bleed" method of decay heat removal. He felt that some demonstration was needed to assure that the feed pumps could deliver enough water at a high enough pressure and that the bleed valves would open and close under operating conditions. Dr. Lewis agreed that the ACRS should confirm the ability to use "bleed and feed" in current PWR designs. It was suggested that an inquiry be made to the NRC Staff as to its intent in this regard.

Meeting With Members of the NRC Staff (Open)

1. Ultrasonic Indications in the Sequoyah Reactor Vessel Nozzles

   Mr. R. M. Gamble from the Materials Engineering Branch of NRR discussed the recently observed ultrasonic indications in the nozzles of the Sequoyah Unit 1 reactor vessel. Ultrasonic indications of cracking have been observed and investigated in a number of reactor vessels since about 1971. Mr. Gamble discussed the cracking mechanisms and the evaluations of the significance of the cracks on vessel performance. The cracks generally result from the weld deposit cladding process used to apply the stainless steel internal surface to the carbon steel vessel and nozzles. In 1971 a single layer of stainless weld metal was deposited using both a pre-and post-clad heat treatment to minimize any tendency to crack. The particular process produced an embrittled zone in the base metal at the interface adjacent to the weld metal. Cracks were observed in this embrittled zone and generally remained within that zone. The cracks occurred when the vessel was given its post clad heat treatment to $1100^{o}F$ and were referred to as reheat cracks. They were extensively studied in 1971. These are very tight cracks about one-eighth inch deep by about one-half inch long that do not extend to the surface of the clad. Dye penetrant techniques generally would not detect these cracks until the cladding had been ground away. In about 1972 it was determined that these cracks did not affect the integrity of the reactor vessels. The Sequoyah vessel was one of those manufactured using this process.

Following the discovery of these cracks, action was taken to reduce the heat input of the cladding process by going to a multi-layer process with proper pre- and post-heat treatments. In 1979 the French discovered another form of cracking that they attribute to hydrogen embrittlement resulting from inadequate post-heat treatment between welding passes. These cracks are about one-quarter inch deep by about three-quarters of an inch long. Reactor vessels manufactured in the U.S. using Reg. Guide 1.43 have not yet shown this type of cracking. The cracks appear to be limited to those vessels manufactured in France or at the Rotterdam Drydock Company where little or no post-heat treatment was used. Because of the cracks found in the foreign made reactor vessels, Westinghouse suggested that the Watts Bar 1 and 2, Sequoyah 1, McGuire 2 and Catawba 1, which were manufactured by Rotterdam Drydock Company, should be inspected. Watts Bar 2 was inspected and no cracking was found but cracks were detected in all nozzles of the Sequoyah 1 vessel. The maximum size of these cracks is about 5/16 ths inch deep by about 5/8 ths inch long. Because the ultrasonic techniques cannot measure actual crack depth, a conservative ratio, based on observations, of depth equals half the length is used. The cracks are within ASME Code allowable limits and no repairs are required. The NRC Staff intends to require inservice inspection of these cracks every ten years to confirm that they are not enlarging.

Mr. Etherington expressed his view that cracks in metal machine components are not new phenomena but that our ability to detect them has become much more sensitive. If the cracks are clearly within the ASME limits he had no concern with the use of the vessel.

The Subcommittee thanked Mr. Gamble for keeping it informed.

2. Recent Blowdown of Primary Coolant at Crystal River Unit 3

Mr. D. G. Eisenhut, Director of the Division of Operating Reactors, briefed a joint session of the Action Plan and Implications Subcommittees on the recent (February 26, 1980) incident at Crystal River Unit 3 in which several thousand gallons of reactor cooling water were released into the containment. The sequence of events began with the reactor operating at 98.6% of rated power and a failure in the +24 volts DC power bus for the non-nuclear instrumentation (NNI) system. There are two channels of NNI (X and Y) with separate DC power supplies, however, isolation of the two channels was not complete. DC power to the X channel was lost which in turn caused a failure of the Y channel. The NNI provides input signals to the Integrated Control System (ICS) which controls, amongst other things, feedwater flow and steam generator water level and provides reactor trip and emergency feedwater start signals. It was postulated that a short circuit through an instrument module on the X +24 volt bus initiated the sequence of events and that the repeated attempts to restore the 24 volt bus eventually burned out the short circuit permitting the 24 volt circuit to be restored. The loss of the 24 volt DC bus caused the power operated relief valve (PORV) to open and the pressurizer spray valve to partially open. Simultaneously the NNI caused the ICS to reduce feedwater flow to the steam generators and there was a loss of indication of major plant parameters including T_{ave} which caused the control rods to withdraw. The reactor tripped on high pressure after the turbine tripped. Following the reactor trip the pressure declined to the point (1500 psi) that the high pressure injection (HPI) started. The reactor coolant pumps were stopped. The operator manually isolated the PORV, the primary system pressure rose and one pressurizer safety valve opened at about 100 psi below its normal set pressure (opened at ~2400 psi instead of 2500 psi) and was kept open by the flow of HPI water. The steam generators were isolated by the ICS steam line rupture logic. The pressure and radiation level in the reactor building went up as the reactor coolant discharge tank inside containment filled and overflowed. Natural circulation was established in the reactor cooling system.

About twenty minutes after the initial loss of DC power, it was restored and recovery operations began. At no time was the core uncovered. A more detailed sequence of events is included as Attachment D.

This failure resulted from a mismatch between the terminals on a printed circuit card and the cabinet into which it was placed. The sequence ended with several thousand gallons of reactor cooling water on the floor of the containment. During the transient the operator followed his procedures that had been updated in accordance with the NRC's Bulletins and Orders that resulted from the TMI-2 accident. The reactor coolant pumps were turned off and the safety injection pumps allowed to run based on the requirements in the NRC Bulletins and Orders.

The integrated control system (ICS) produced many erroneous signals as a result of the loss of the DC power supplies. However, the reactor protection system was unaffected and performed its functions correctly.

It was noted that a Shift Technical Advisor was present in the control room but was never called upon nor needed for advice. It was also noted that an NRC resident inspector and several (eight) other NRC personnel were on site, some in the control room, during this event.

As a result of this transient, the NRC Staff has asked licensees for more information regarding failure modes and effects of non-safety grade instruments and controls (including power supplies) and how this could feed into safety systems.

3. Draft 3 NRC Action Plans Developed as a Result of the TMI-2 Accident (NUREG-0660)

Mr. Etherington advised Mr. Mattson that this portion of the Subcommittee meeting should be devoted to only those items in the Action Plan that pertain to nuclear power plants that would be expected to receive their operating licenses in the near-term (NTOL).

Mr. Mattson, Director of the Division of Systems Safety, presented an overview of Draft 3 of the Action Plan and an outline of how he proposed to cover the material to be considered during this meeting. Mr. Purple, from the Action Plan Steering Group, reviewed the history of the development of the Action Plan and how ACRS recommendations were factored into the plan.

Table 1 in Draft 3 is essentially an annotated index of all action plan items and shows the decision status of each item. Decision Group A includes those things that have essentially been decided by the Commission. It includes all of the Bulletins and short term lessons learned as well as approved operator licensing reforms. Decision Group B incorporates those items that can be well enough defined so that schedules can be established for their implementation.

Decision Group C includes those items that the Commission is not prepared to make a decision on and that will have to be studied further.

Decision Group D includes those items that are really part of basic programs of the agency and will be handled as part of the normal business of the agency or can be deferred until 1982 or beyond.

Table 1 also indicates an implementation schedule such as "fuel load," "full power operation" or a deadline date.

Mr. Mattson called the Subcommittee's attention to Appendix A and Table A.1 which describe the near-term operating license (NTOL) requirements. Appendix B describes the method of assigning priorites to the various tasks within the Action Plan. Appendix C addresses the recommendations and requirements based on IE Bulletins and Orders and on Commission Orders. There was a good deal of discussion regarding implementation schedules and the role of the Steering Group and the ACRS in evaluating the judgement of the NRC Staff with regard to various proposed actions and their implementation.

The comments of the Atomic Industrial Forum review group (based on Draft 2) have not been specifically factored into the current Draft 3. However, the evolution of Draft 3 did result in some of the AIF concerns being resolved. It was noted that on the day following this Subcommittee meeting (March 6, 1980) the full ACRS would hear a presentation from the AIF on the action plan. Dr. Mattson also noted that, in some cases, the AIF cost estimates were more accurate than the earlier estimates of the NRC Staff. However, the major cost element in the AIF estimate was the cost of replacement power during the period that the NRC Staff would require plants to remain shutdown to make changes. The NRC Staff indicated that it would probably be more flexible than the AIF predicted and the costs would not nearly approach the $32 B estimated by the AIF for delays in completion.

Roughly, the AIF estimates the cost per NTOL item per plant to be about $2.4 $\overline{M}$ while the NRC estimate is $1.5 $\overline{M}$. This is just for comparison purposes and the costs per specific item vary greatly.

Mr. Russell described the various methods of prioritization used in the various drafts of the Action Plan. The current method of point assignment is described in Appendix B. He noted that the priorities assigned by the AIF included one group they thought should be eliminated from the Action Plan. Table B.2 includes the AIF priority as well as an indication of the date of the ACRS letter which spoke to the topic. Three ACRS concerns (overfilling the steam generators, seismic implications, and quantitative safety goals) are not addressed in the current draft of the Action Plan but might be considered to be subparts of proposed actions.

Mr. Mattson noted that not all items in the Action Plan will be implemented in FY '80 and '81, some will be deferred until FY '82 and beyond. He described the inverse priority system applied to the current NRC operating plan items to determine which of those could be deleted or deferred to make room for the Action Plan items. Attachment E illustrates the proposed reprogramming of NRR manpower to accomplish the NTOL portion of the Action Plan. It is estimated that the first ten items will have to be dropped in order to do the NTOL items and all twenty would have to be dropped if all of the actions proposed in Draft 1 were to be done.

In spite of the problems that had to be addressed by the NRC Staff following the TMI-2 accident, NRR has continued to work on the twenty unresolved safety issues. The prioritization used to identify the top twenty unresolved safety issued from the 133 generic issues is shown in Attachment F.

Mr. Mattson went through the list of Near-Term Operating License (NTOL) items as listed in Table A.1. He pointed out that they were broken down into five catagories. The first catagory contains those requirements that have not previously been issued and are in addition to the short term lessons learned; the second catagory is the short term lessons learned; the third catagory is the actions that affect NRC organization, policy, or procedures; the fourth contains recommendations coming from the NRC Special Inquiry Group (Regovin Report); and the fifth is a list of items dropped from the list proposed in Draft 2. Mr. Mattson suggested that the Subcommittee concentrate on the first and fourth catagories since the second is essentially fait accompli and the third applies only to the NRC itself

Dr. Kerr noted that the NRC would evaluate the utility's management capability based on draft internal NRC criteria. He asked about the availability of these criteria and expressed the position that the utility should know the criteria it is to be judged against. A copy of the draft criteria was provided.

The Subcommittee considered each item in Part 1 of Table A.1 and did not suggest the deletion of any item. The Subcommittee suggested that the architect-engineer be included in the review of plant procedures to assure that systems are used as intended. It was also suggested that emergency procedures be based on symptoms an operator would observe and actions he must take to relieve those symptoms. Another suggestion was to take advantage of the low power test program to obtain R&D type data on a full scale plant.

With regard to Part 2 of the NTOL list, Item 15, Mr. Lipinski expressed his concern that the hydrogen recombiner penetrations should take suction from the high point in the containment rather than some intermediate point.

Mr. Mattson said that it was the NRC Staff's position that the NTOL list was a necessary and sufficient set of requirements to permit the resumption of licensing, including full power licensing.

Mr. O'Reilly described how a NRC Staff team visited various NTOL and operating plants to discuss the NTOL requirements to determine if there would be any unexpected adverse consequences of the proposed actions. The discussions included not only plant managers but also the licensed operators that would have to live with the proposed actions. The primary negative impact identified was that of distracting the currently limited pool of talent away from current operating problems to address the NTOL items. As a result of these discussions, some of the deadlines established earlier were relaxed in order to not overtax the utilities resources.

Dr. Kerr pointed out that the change in PORV and Scram pressure set points has reduced the number of PORV openings and increased the number of Scrams. He was not sure that the increased safety resulting from the reduced frequency of opportunity for the PORV to stick open was matched by the increased opportunity for ATWS because of the increased frequency of challenge of the Scram system. Mr. Mattson thought the Integral Reliability Evaluation Program (IREP) would determine if the changes resulted in a net improvement in safety.

The Subcommittee discussed the schedule and presentations to be made before the full ACRS on the following day (March 6, 1980). The Subcommittee adjourned at 7:15 P.M.

A complete transcript of the meeting is on file at the NRC Public Document Room at 1717 H St., N.W., Washington, D.C. or can be obtained from the Internation Verbatim Reporters, Inc., Suite 107, 449 South Capitol Street, S.W. Washington, D. C. 20002, 202/484-3550.

JCK

(a) Lighting fixtures on the side of the equipment would "blind" the operator and nearby miners or require constant adjustment to changes in illumination; fixtures would be sheared off or broken, increasing the likelihood of more serious equipment failure, wedging, jamming or upset. Also, as lighting fixtures on the side or top are sheared off, roof bolts, cross beams and straps will be sheared off, thereby damaging or destroying roof support.

(b) Installation of stationary lighting equipment would similarly impair the operators' and nearby miners' vision. It would also create additional heat in the confiningly small areas in which the miners must work.

3. For these reasons, the petitioner requests a modification of the application of the standard to its mine.

Request for Comments

Persons interested in this petition may furnish written comments on or before March 20, 1980. Comments must be filed with the Office of Standards, Regulations and Variances, Mine Safety and Health Administration, Room 627, 4015 Wilson Boulevard, Arlington, Virginia 22203. Copies of the petition are available for inspection at that address.

Dated: February 11, 1980.

Frank A. White,

Director, Office of Standards, Regulations and Variances.

[FR Doc. 80-4105 Filed 2-15-80; 8:45 am]

BILLING CODE 4510-43-M

NUCLEAR REGULATORY COMMISSION

Advisory Committee on Reactor Safeguards, Subcommittee on Three Mile Island, Unit 2 Accident Implications; Meeting

The ACRS Subcommittee on the Three Mile Island, Unit 2 Accident Implications will hold a meeting on March 5, 1980 in Room 1046, 1717 H St., NW., Washington, DC 20555 to consider the potential installation of molten core crucibles under the Indian Point 2 & 3 and the Zion 1 & 2 Reactors. Notice of this meeting was published January 22, 1980.

In accordance with the procedures outlined in the Federal Register on October 1, 1979, (44 FR 56408), oral or written statements may be presented by members of the public, recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Subcommittee, its consultants, and Staff. Persons desiring to make oral statements should notify the Designated Federal Employee as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements.

The agenda for subject meeting shall be as follows: Wednesday, March 5, 1980, 8:30 a.m. until the conclusion of business.

The Subcommittee may meet in Executive Session, with any of its consultants who may be present, to explore and exchange their preliminary opinions regarding matters which should be considered during the meeting.

At the conclusion of the Executive Session, the Subcommittee will hear presentations by and hold discussions with representatives of the NRC Staff, the Power Authority of the State of New York, the Consolidated Edison Co. of New York, Inc., the Commonwealth Edison Co., their consultants, and other interested persons.

In addition, it may be necessary for the Subcommittee to hold one or more closed sessions for the purpose of exploring matters involving proprietary information. I have determined, in accordance with Subsection 10(d) of the Federal Advisory Committee Act (Pub. L. 92-463), that, should such sessions be required, it is necessary to close these sessions to protect proprietary information. See 5 U.S.C. 552b(c)(4).

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the cognizant Designated Federal Employee, Mr. Richard K. Major (telephone 202/634-1414) between 8:15 a.m. and 5:00 p.m., EST.

Background information concerning items to be discussed at this meeting can be found in documents on file and available for public inspection at the NRC Public Document Room, 1717 H Street, N.W., Washington, DC 20555 and at the Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Street, Harrisburg, PA 17126 (regarding Three Mile Island), the White Plains Public Library, 100 Maritime Avenue, White Plains, New York 10601, (regarding Indian Point), and the Zion-Benton Public Library, 2600 Emmaus Avenue, Zion, IL 60099 (regarding Zion).

Dated: February 13, 1980.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 80-5[illegible] Filed 2-15-80; 8:45 am]

BILLING CODE 7590-01-M

Advisory Committee on Reactor Safeguards, Ad Hoc Subcommittee on Three Mile Island, Unit 2 Accident Action Plan; Meeting

The ACRS Ad Hoc Subcommittee on the Three Mile Island, Unit 2 Accident Action Plan will hold a meeting on March 5, 1980 in Room 1167, 1717 H St., NW, Washington, DC 20555 to consider Draft 3 of the NRC "Action Plans for Implementing Recommendations of the President's Commission and Other Studies of the Three Mile Island, Unit 2 Accident."

In accordance with the procedures outlined in the Federal Register on October 1, 1979 (44 FR 564[illegible]), oral or written statements may be presented by members of the public, recordings will be permitted only during those portions of the meeting when a transcript is being kept, and questions may be asked only by members of the Subcommittee, its consultants, and Staff. Persons desiring to make oral statements should notify the Designated Federal Employee as far in advance as practicable so that appropriate arrangements can be made to allow the necessary time during the meeting for such statements.

The agenda for subject meeting shall be as follows: Wednesday, March 5, 1980, 8:30 a.m. until the conclusion of business.

The Subcommittee may meet in Executive Session, with any of its consultants who may be present, to explore and exchange their preliminary opinions regarding matters which should be considered during the meeting.

At the conclusion of the Executive Session, the Subcommittee will hear presentations by and hold discussions with representatives of the NRC Staff, the nuclear industry, various utilities, and their consultants, and other interested persons.

In addition, it may be necessary for the Subcommittee to hold one or more closed sessions for the purpose of exploring matters involving proprietary information. I have determined, in accordance with Subsection 10(d) of the Federal Advisory Committee Act (Pub. L. 92-463), that, should such sessions be required, it is necessary to close these sessions to protect proprietary information. See 5 U.S.C. 552b(c)(4).

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the cognizant Designated Federal Employee, Mr. John C. McKinley

ATTACHMENT A

(telephone 202/634-3265) between 8:15 a.m. and 5:00 p.m., EST.

Background information concerning items to be discussed at this meeting can be found in documents on file and available for public inspection at the NRC Public Document Room, 1717 H Street, N.W., Washington, DC 20555 and at the Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Street, Harrisburg, PA 17126.

Dated: February 13, 1980.

John C. Hoyle,

Advisory Committee Management Officer.

[FR Doc. 80-5100 Filed 2-15-80; 8:45 am]

BILLING CODE 75-90-01-M

[NUREG/CR-1263]

Compilation of State Laws and Regulations That Deal With Radioactive Materials in Transport

The Office of State Programs and the Office of Standards Development, Division of Engineering Standards have issued a compilation of State laws and regulations that deal with radioactive materials in transport. It is current as of December 28, 1979. Local government initiatives are not included.

A copy is available for review at the Nuclear Regulatory Commission (NRC) Public Document Room, 1717 H Street, NW., Washington, D.C. A single copy of NUREG/CR-1263 will be provided free of charge, while the supply lasts, upon written request of a full participant in an ongoing NRC proceeding. The request must identify the requester as a participant and should be addressed to Director, Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Copies may be purchased at current rates from the GPO Sales Program, Division of Technical Information and Document Control, NRC, Washington, D.C. 20555, and the National Technical Information Service, Springfield, Virginia 22161.

Dated at Bethesda, Maryland this 12th of February, 1980 for the Nuclear Regulatory Commission.

G. Wayne Kerr,

Acting Director, Office of State Programs.

[Doc. 80-5098 Filed 2-15-80; 8:45 am]

BILLING CODE 7590-01-M

[Dockets Nos. 50-275 and 50-323]

Pacific Gas & Electric Co.; Diablo Canyon Nuclear Power Plant, Units 1 and 2; Order Extending Construction Completion Dates

Pacific Gas and Electric Company is the holder of Construction Permit Nos. CPPR-39 and CPPR-69 issued by the Atomic Energy Commission [1] on April 23, 1968 and December 9, 1970, respectively, for construction of the Diablo Canyon Nuclear Power Plant, Units 1 and 2, presently under construction at the Company's site in San Luis Obispo County, California.

On November 26, 1979, Pacific Gas and Electric Company filed a request for extensions of the completion dates for Units 1 and 2 based on the time needed to comfortably accommodate the Commission's announcement of a pause in issuing operating licenses to the spring of 1980.

As a result of the accident at the Three Mile island, Unit 2 Nuclear Power Plant, the Nuclear Regulatory Commission (NRC) conducted an extensive investigation of potential design deficiencies in the plant, plant operator response to the accident, operator errors and/or misinterpretation of plant instrumentation, and all other aspects of the accident which might lead to information that would improve the safety of nuclear power plants. The findings of this investigation and from the comprehensive study of the President's Commission showed that certain plant modifications and improvements in utility organizational structure were required to improve the safety of nuclear power plant operation and to upgrade the capability of utilities to cope with a severe accident. At this time the NRC staff is preparing an Action Plan for Commission review and approval that will include new or improved safety objectives, the detailed criteria for their implementation and the various implementation deadlines. The goal is to have the Action Plan available for Commission review and approval by mid-February 1980. When the Commission approves the Action Plan, the requirements will be submitted to the utilities for implementation. In order to accommodate a reasonable schedule needed to complete the safety evaluation regarding the applicant's response to the Action Plan requirements and the subsequent licensing actions related to these new safety requirements, the Nuclear Regulatory Commission staff concluded that the requested completion dates of construction for Unit 1 and for Unit 2 should be extended to September 30, 1980 and March 31, 1981, respectively.

This action involves no significant hazards consideration: good cause has been shown for the delays; and environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with the extension; and the requested extension is for a reasonable period, the bases for which are set forth in a staff evaluation of request for extension.

[1] Effective January 20, 1975, the Atomic Energy Commission became the Nuclear Regulatory Commission and Permits in effect on that day were continued under the authority of the Nuclear Regulatory Commission.

For further details with respect to this action, see (1) the applicant's request for extension of the construction permit completion date for Diablo Canyon, Units 1 and 2 dated November 26, 1979, and (2) the staff's related evaluation, all of which are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555 and at the Local Public Document Room located at the California Polytechnic State University Library, Document and Maps Department, San Luis Obispo, California 93407.

It is hereby ordered that the latest completion date for CPPR-39 is extended from December 31, 1979 to September 30, 1980 for Unit 1 and the latest completion date for CPPR-69 is extended from February 29, 1980 to March 31, 1981 for Unit 2.

Date of Issuance: February 2, 1980.

For the Nuclear Regulatory Commission.

D. F. Ross, Jr.,

Acting Director, Division of Project Management, Office of Nuclear Reactor Regulation.

[FR Doc. 80-5102 Filed 2-15-80; 8:45 am]

BILLING CODE 7590-01-M

Three Mile Island; Determination of Extraordinary Nuclear Occurrence

The Commission hereby extends the time in which to determine whether the accident at Three Mile Island constitutes an "extraordinary nuclear occurrence" until March 14, 1980.

Dated this 12th day of February 1980, at Washington, D.C.

For the Commission.

Samuel J. Chilk,

Secretary of the Commission.

[FR Doc. 80-5150 Filed 2-15-80; 8:45 am]

BILLING CODE 7590-01-M

OFFICE OF MANAGEMENT AND BUDGET

Agency Forms Under Review

Background

February 13, 1980.

When executive departments and agencies propose public use forms, reporting, or recordkeeping requirements, the Office of Management and Budget (OMB) reviews and acts on those requirements under the Federal

TENTATIVE SCHEDULE
FOR
AD HOC SUBCOMMITTEE MEETING ON NRC ACTION PLANS
DEVELOPED AS A RESULT OF THE TMI-2 ACCIDENT
MARCH 5, 1980

8:30 A.M. EXECUTIVE SESSION (OPEN)

1. Near-term Operating License (NTOL) Requirements (Draft #2 of Action Plan and Feb. 6 letter from Dircks to Commissioners).

2. Chairman Ahearne's request regarding sufficiency of NTOL items.

9:30 A.M. Meeting with NRC Staff - (J. Knight et al).

1. Recently observed ultrasonic indications in nozzles of the Sequoyah reactor vessel.

10:00 A.M. Meeting with NRC Staff - (D. Eisenhut et al).

1. Implications of recent primary system blowdown at Crystal River.

10:30 A.M. Meeting with NRC Staff - (R. Mattson et al)

1. Any changes in NTOL requirements resulting from Draft #3 or from the Sequoyah review.

2. Possible additions to the program.

   a. A commitment to the interim reliability evaluation program (IREP).

   b. A commitment to studies of containment modifications which would allow the containment to accommodate accidents beyond the DBA.

   c. A commitment to studies of possible design changes which would improve reliability of systems needed for safe shutdown; accounting for the possibility of multiple failures in non-safety grade equipment.

   d. Feed and bleed decay heat removal capability including detailed analyses and operating procedures.

ATTACHMENT B

3. Possible deletions from the program

   a. Reconsider the current requirement to trip the reactor coolant pumps following a small break LOCA.

   b. Reconsider the criteria for securing (throttling) the HPI following HPI initiation on low pressure.

12:30 P.M. ***** LUNCH *****

1:30 P.M. Resume Meeting With NRC Staff

5:00 P.M. ADJOURN

LIST OF DOCUMENTS AVAILABLE TO
ACRS AD HOC SUBCOMMITTEE ON NRC ACTION PLAN
March 5, 1980

1. "NRC Action Plans Developed as a Result of the TMI-2 Accident" Draft 3 NUREG-0660 3/5/80.

2. "Tentative Schedule for Ad Hoc Subcommittee Meeting on NRC Action Plans Developed as a Result of the TMI-2 Accident, March 5, 1980."

3. Memo to H. Etherington from J. C. McKinley dated February 29, 1980 subject "Three Mile Island Action Plan Subcommittee Meeting March 5, 1980" with attachments.

4. Package of background material which included:

   a. Draft memo, dated February 22, 1980 to Dr. Plesset from John G. Stampelos, ACRS Fellow.

   b. Memorandum dated February 22, 1980 for Milton Plesset from Roger J. Mattson, TMI-2 Action Plan Steering Group.

   c. ACRS report dated January 15, 1980 to John F. Ahearne, Subject: Draft NUREG-0660.

   d. ACRS report dated February 11, 1980 to John F. Ahearne, Subject: NUREG-0660 Draft 2.

   e. Memorandum dated February 19, 1980 for Chairman, ACRS from John Ahearne, Subject: Commission Use of ACRS Views.

   f. Memorandum dated February 22, 1980 for Dr. Milton S. Plesset from Victor Gilinsky.

   g. ACRS report dated December 11, 1979 to John F. Ahearne, Subject: Interim Low Power Operation of Sequoyah Nuclear Power Plant, Unit 1.

   h. Memorandum dated February 22, 1980 for ACRS members from R. Savio, ACRS Staff Engineer, Subject: "Status of NRC Actions on the Sequoyah Review," (with attachments).

   i. Memorandum dated February 25, 1980 for ACRS members from R. Savio, ACRS Staff Engineer, Subject: "ACRS Review of the Action Plan, NTOL List, a d the Sequoyah OL Application."

   j. Memorandum dated Feb. 6, 1980 for Chairman Ahearne, Commissioners Gilinsky, Kennedy, Hendrie, Bradford from William J. Dircks, Subject: "Staff Review of the Report by the NRC Special Inquiry Group on the Accident at Three Mile Island."

   k. Atomic Industrial Forum letter to Harold Denton and signed by Byron Lee, Jr. regarding the Action Plan priorities and resources.

5. Memorandum dated March 4, 1980 for M.S. Plesset from John G. Stampelos, ACRS Fellow Subject: "Reviews of NRC TMI Action Plan (NUREG-0660; Draft 2; January 23, 1980) and Examples of Possible Rush to Judgement Items" (with attachments)

6. Handout material (10 pages) by R. M. Gamble on the underclad cracking in reactor vessel nozzles (Sequoyah Unit 1 reactor vessel).

7. Handout material (20 pages) by D. Eisenhut et al on the recent event at Crystal River Unit 3.

8. Handout material (4 pages) by R. Purple on history of development of NUREG-0660.

9. Handout material (17 pages) by R. J. Mattson on prioritization and allocation of resources relating to NUREG-0660.

10. NRC Draft (2/25/80) Criteria for Utility Management and Technical Competence.

SEQUENCE (AS OF 2300 3/1/80)

26 February Transient CR-3

EVENT SYNOPSIS

At 14:23 on February 26, 1980 Crystal River -3 Nuclear Station experienced a reactor trip from approximately 100% full power. A synopsis of key events and parameters was obtained from the plant computer's post-trip review and plant alarm summary, the sequence of events monitor, control room strip charts, and the Shift Supervisor's log.

The reactor was operating at approximately 100 % full power with Integrated Control System (ICS) in automatic. No tests were in progress and minor maintenance was being performed in the Non-Nuclear Instrumentation (NNI) cabinet "Y".

Time	Event	Cause/Comments
14:23:00	The following is a summary of plant conditions prior to the trip Flux 98.6% RC Pressure 2157 psig PZR level 202 inches MU tank level 71 inches T_H "A" 599°F. T_H "B" 600°F. T_C "A" 557°F. T_C "B" 556°F. RC Flow "A" 73 X 10^6 lbs/hr RC Flow "B" 73 X 10^6 lbs/hr Letdown Flow 48 gpm OTSG "A" lvl (OP) 67% OTSG "B" lvl (OP) 65% OTSG "A" FRLV 242 inches OTSG "B" FRLV 254 inches OTSG "A" Pressure 911 psig OTSG "B" pressure 909 psig Main Steam Pressure 894 psig Main Steam Temp. 589°F. Condenser Vacuum 1.76 Generated MW 834 DFT level 12.7 ft. Feed Flow "A" 5 X 10^6 lbs/hr Feed Flow "B" 5 X 10^6 lbs/hr Feed Pressure "A" 970 psig Feed Pressure "B" 968 psig	
14:23:21	+24 Volt Bus Failure (NNI power loss "X" supply)	Cause still unknown. Apparently, the positive 24 VDC bus shorted dragging the bus voltage down to a

ATTACHMENT D

Time	Event	Cause/Comments
		low voltage trip condition. There s a built-in ¼ to ½ second delay at which time all power supplies will trip. There was no trip indication on negative (-) voltage. This event was missed by the annunciator. Following the NNI power failure, much of the control room indication was lost. Of the instrumentation that remained operable transient conditions made their indication questionable to the operators.
14:23:21	PORV and Spray Open	When the positive 24 VDC supply was lost due to the sequence discussed above the signal monitors in NNI changed state causing PORV/Spray valves to open. The PORV circuitry is designed to seal in upon actuation and did so. The resultant loss of the negative 24 VDC halted spray valve motor operator and prevented PORV seal in from clearing on low pressure. It is postulated that the PORV opened fully and the spray valve stroked for approximately ½ second. The 40% open indication on spray valve did not actuate, therefore, the spray valve did not exceed 40% open.
14:23:21	Reduction in Feedwater	As a result of the "X" power supply failure many primary plant control signals responded erroneously. Tcold failed to 570°F (normal indication was 557°F) producing several spurious alarms Tave failed to 570°F (decreased). The resultant Tave error modified the reactor demand such that control rods were withdrawn to increse Tave and reactor power. The power increase was terminated at 103% by the ICS and a "Reactor Demand High Limit" alarm was received. Thot failed to 570°F (low) and RC flow failed to 40 X 10^6 lbs/hr in each loop (low). Both these failures created a BTU alarm and limit on feedwater which reduced feedwater flow to both OTSG's to essentially zero. Turbine Header Pressure failed to 900 psig (high) which caused the turbine valves to open slightly to

Time	Event	Cause/Comments
		regulate header pressure thus increasing generated megawatts. These combined failures resulted in a loss of heat sink to the reactor initiating an excessively high RC pressure condition.
14:23:35	Reactor Trip/Turbine Trip	Rx trip caused by high RCS pressure at 2300 ps: Turbine was tripped by the reactor.
14:24:02	Hi Pressure Inj. Req. (Flag)	This was a computer printout and indicates <50° subcooling.* See attached graph of RC Pressure/Temp. vs. Time. This graph is based on Post Trip data and actual incore thermocouple data. From the reactor trip point (14:2: to 14:33, core exit temperature data was obtained by extrapolation and calculated data. This is supported by two alarm data points plotted at 18° and 21° of subcooling during this period from the computer. It is important to note that lowest level of subcooling was 8°F for a very short period of time. *NOTE: This computer program was initiated as a result of the TMI incident.
14:24:02	Loss Of Both Condensate Pumps	Suspect condensate pump tripped due to high DFT level. This is verified by ???? printed by computer, indicating the level instrument was over ranged as well as a low flow indication in the gland steam condenser as als indicated by computer.
14:25:50	PORV Isolated	At this time a high RC Drain Tank level alarm was received. This was resultant from the PORV remaining open and was positive indicatio that the PORV was open. At this time, the operator closed the PORV block valve due to RCS pressure decreasing and high RCDT level.
14:26:41	HPI Auto Initiation	HPI initiated automatically due to low RCS pressure of 1500 psig. The low pressure condition was resultant from the PORV remainin full open while the plant was tripped. Full HPI was initiated with 3 pumps resulting in approximately 1100 gpm flow to the RCS. At this time, all remaining non-essential R.B. isolation valves

Time	Event	Cause/Comments
		were closed per TMI Lessons Learned Guidelines
14:26:54	RC Pumps Shutdown	Operator turned RC pumps off as required by the applicable emergency procedure and B & W small break guidelines.
14:27:20	RB Pressure Increasing	This is first indication that RCDT rupture disc had ruptured. RB pressure increase data was obtained from Post Trip Review and Strip Chart indication.
14:31:32	RB Pressure High	This alarm was initiated by 2 psig in RB. This is attributed to steam release from RCDT. Code safeties had not opened at this time based upon tail pipe temperatures recorded at 14:32:03 (Computer).
14:31:49	OSTG "A" Rupture Matrix Actuation	This occurred due to <600 psig in OTSG "A". The low pressure was caused by OTSG "A" boiling dry which was resultant from the BTU limit and failed OTSG level transmitter. This resulted in the closure of all feedwater and steam block valves which service OTSG "A".
14:31:59	Main Feedwater Pump 1A Tripped	Caused by suction valve shutting due to matrix actuation in previous step.
14:32→14:41	ES A/B Bypass	Manually bypassed and HPI balanced between all 4 nozzles (Total flow approximately 1100 gpm -small break operating guidelines).
14:32:35	Started Steam Driven Emergency Feedwater Pump	Started by operator to ensure feedwater was available to feed OTSG's.
14:33	Core Exit Temp. Verified	The core exit incore thermocouples indicated the highest core outlet temperature value was 560°F. RCS pressure was 2353 psig atthis time therefore, the subcooling margin at this time was <u>100°F</u>. Minimum subcooling margin for the entire transient was <u>8°F</u>. It is postulated that some localized boiling occurred in the core at this point as indicated by the self powered neutron detectors.
14:33-14:[illegible]	Started Motor Driven Emergency Feedwater Pump	Same discussion as "Started Steam Driven Emergency Feedwater Pump."
14:33:30	RC Pressure High (2395 psig)	At this point, pressurizer is solid and code safety lifts (RCV-8). This is the highest RCS pressure as recorded on Post Trip Review. Apparently, RCV-8 lifted early due to seat

Time	Event	Cause/Comments
		leakage prior to the transient and RCV-9 did not lift.
14:34:23	RB Dome Hi Rad Level	RMG-19 alarmed at this point. Highest level indicated during course of incident was 50 R/hr. High radiation levels in RB caused by release of non-condensable gases in the pressurizer and coolant.
14:35:33	Attempted NNI Repower Without Success	This resulted in spikes observed on de-energized strip charts.
14:36:50	Computer Overload	Caused by overload of buffer. Resulting in no further computer data until buffer catches up with printout.
14:38:15	FWV-34 Closed	This valve was closed to prevent overfeeding OTSG "B" beyond 100% indicated Operating Range.
14:44:12	NNI Power Restored Successfully	NNI was restored by removing the X-NNI Power Supply Monitor Module. This allowed the breakers to be reclosed. At this time, it was observed that the "A" OTSG was dry, the pressurizer was solid (Indicated off scale high). RC outlet temperature indicated 556°F (Loop A & B average), and RC average temperature indicated 532°F (Loop A & B). The highest core exit thermocouple temperature at this time was [illegible]°F. RSC pressure was 2400 psig (saturation temp. at this pressure is 662°F.). This data verifies natural circulation was in progress and the plant subcooling margin was [illegible]°F. (based on core exit thermocouples).
14:44:31	RB Isolation and Cooling Actuation	At this time, RB pressure increased to 4 psig and initiated RB Isolation. The operator verified all immediate actions occurred properly for HPI, LPI, and RB Isolation and Cooling. The increasing RB pressure was resultant from [illegible] passing HPI at this time.
14:46:10	Bypassed HPI, LPI and RB Isolation and Cooling	These "ES" systems were bypassed at this time to again balance HPI flow and restore cooling water to essential auxiliary equipment (i.e. RCP's, letdown coolers, CRDM's etc. .

Time	Event	Cause/Comments
14:51:57	Rupture Matrix Actuation on OTSG-B	The actuation was resultant from a degradation of OTSG-B pressure. Cold emergency feed was being injected into the OTSG at this time. This matrix actuation isolated all feedwater and steam block valves to the B-OTSG and tripped the "B" main FW pump. Both Emergency FW pumps were already in operation at this time. B-OTSG level at this time was 70% (Operation Range).
14:52	HPI Throttled and RCS Pressure Reduced to 2300 psig	At this time, the maximum core exit thermocouple temperature was 515°F, RCS pressure was 2390 psig. Therefore, the subcooling margin was 147°F. Natural circulation was in effect as verified previously. All conditions had been satisfied to throttle HPI. Therefore, flow was throttled down to approximately 250 gpm to reduce RCS pressure to 2300 psig in order to attempt to reduce the flow rate through RCV-8 and into the RB.
14:53	Reestablished Letdown	At this time, the operator was attempting to establish RCS pressure control via normal RC makeup and letdown.
14:56	Opened MU Pump Recirc. Valves	This was done to assure the MU pumps would have minimum flow at all times to prevent possible pump damage.
14:56:43	Bypassed the A-OTSG Rupture Matrix and Reestablished Feed to the A-OTSG	Feedwater was slowly admitted to the A-OTSG which was dry up to this point. Feedwater was admitted through the Auxiliary FW header via the EFW bypass valves. The feedrate was very slow in order to minimize thermal shock to the OTSG and resultant depressurization of the RCS. RCS pressure control was very unstable at this time. It is postulat that some localized boiling occured in core at this point as indicated by self neutron detectors.

Time	Event	Cause/Comments
14:57:09	Bypassed the B-OTSG Rupture Matrix	This was done to regain FW control of the B-OTSG. Level was still high in this OTSG (approximately 65% Operating Range). Therefore, feed was not necessary at this time. The Main Steam Isolation valves were open in preparation for bypass valve operation (when necessary).
14:57:15	Established RC Pump Seal Return	This was done in preparation for a RCP start (when necessary) and to minimize pump seal degradation.
15:00:09	Reestablished Level In A-OTSG	This verified feedwater was being admitted to the OTSG and made it available for core cooling via natural circulation. Feed to this generator was continued with the intent of proceeding to 95% on the Operating Range.
15:00:09	77°F Subcooled "A" Loop	This value was based upon "A" RCS loop parameters at this time. The "A" loop was being cooled down at this time by the A-OTSG fill and the operator was attempting to equalize loop temperatures.
15:15	23°F Delta-T/Manned the Technical Support Center	At this time, loop temperatures were nearing equalization. This delta-T was calculated from loop A & B T_C's and core exit thermocouples.
15:17	Declared Class "B" Emergency	This was done based on the fact there was a loss of coolant through RCV-8 in the containment and HPI had been initiated. All non-essential CR# 3 personnel were directed to evacuate and contact off-site agencies be-gan. Survey team was sent to Auxiliary Building
15:19	Opened Emergency FW Block to B-OTSG	At this point the A-OTSG level was increasing and the decision was made to commence filling the B-OTSG simultaneously. The intent was to go 95% on both OTSG's without exceeding RCS cooldown limits (100°F/hr) while maintaining RCS pressure control.

Time	Event	Cause/Comments
15:26	Lo Level Alarm in Sodium Hydroxide Tank	This was resultant from the tank supply valve opening when the 4 psig RB isolation and cooling signal actuated. The sodium hydroxide was released to both LPI trains. Sodium Hydroxide was admitted to the RCS via HPI from the BWST. (Approximately 2 ppm injected into the RCS.)
15:50	Terminated HPI	At this time, all conditions had been satisfied (per small break operating guidelines) to terminate HPI. RCS pressure control had been established using normal makeup and letdown. HPI was terminiated and essentially all releases to the RB were discontinued.
16:00	Commenced Pressurizer Heatup	At this time, RCS pressure and temperature were well under control. Natural circulation was functioning as designed (approximately 23° delta-T). RCS temperature was being maintaine at approximately 450°. RCS pressure was appro imately 2300 psig. The decision was made at this point to commence pressurizer heatup in preparation to re-establish a steam space in the pressurizer.
16:07	Survey Team Report	The Emergency Survey Team reported no radiatio survey results taken offsite were above background.
16:08 :04	Shutdown Steam Drive Emergency FW Pump	The motor driven Emergency FW pump was running therefore, the steam driven pump was not neede The plant remained in this condition for approximately 2 hours, while heating up the press urizer to saturation temperature for 1800 psig
16:15	Press Release	Media was notified of plant status.
18:05	Established Steam Space Pressurizer	At this point, pressurizer temperature was approximately 610°F. Pressurizer level was brought back on scale by increasing letdown. From this point pressurizer level was reduced to normal operating level and normal pressure was established via pressure heaters.
18:30	Terminated Class B Emergency	State and Federal Agencies notified.

Time	Event	Cause/Comments
21:07	Forced Flow Initiated in RCS	The decision was made to re-establish forced flow cooling in the RCS at this time. B&W and NRC were consulted. RCP-1B and 1D were started. At this point, RCS parameters were stabilized and maintained at RC pressure-2000 psig, RCS temperature-420°F. Pressurizer level-235 inches. The plant was considered in a normal configuration.

Proposed Order of Reprogramming of NRR Manpower for Approved TMI-2 Actions

Brief Description of Reprogramming Action	Estimated Manpower Freed FY80 (7 mo) PMY	FY80 (7 mo) Cumu	FY81 (12 mo) PMY	FY81 (12 mo) Cumu
1. Defer work on Early Site Reviews thru FY 81	5	5	1	1
2. Eliminate Advanced Reactors work thru FY81	2	7	3	4
3. Defer standard plant reviews thru FY81	6	13	16	20
4. Substantially reduce NRR participation in standards development thru FY 81	3	16	4	24
5. Reduce NRR support to other offices to minimum level thru FY 81	5	21	12	36
6. Reduce work on revisions to Standard Review Plan and eliminate work on licensing improvements in FY 81	0	21	10	46
7. Defer work on CP applications thru FY 81, except those with complete SER	13	34	13	59
8. Substantially reduce effort on audit calculations thru FY 81	4	38	4	63
9. Defer work on non-critical topical reports thru FY 81	2	40	4	67
10. Defer all staff work on generic issues other than "Unresolved Safety Issues" thru FY 80, and reduce level of staff manpower on these issues to 40% of budget for FY 81	42	82	31	98
NTOL Reprogramming Line				
11. Reduce level of staff manpower on OL reviews by 50% for remainder of FY 80 and by 30% for FY 81 (will further delay issuances of scheduled reviews for FY 80 and 81)	18		26	
12. Reduce Post-CP work by one half	4		-	
13. Reduce level of staff manpower on OR routine actions by 10% for remainder of FY 80	9		-	
14. Stop all remaining work on OL reviews	17		-	
15. Reschedule USI as needed to reduce manpower by 40% for remainder of FY 80	5		-	
16. Reschedule SEP as needed to reduce manpower by 40% for remainder of FY 80	5		-	
17. Eliminate all Training for remainder of FY 80	4		-	
18. Reduce SEP by additional 40% (leaves only care-taker function) for remainder of FY 80	5		-	
19. Further reschedule USI to reduce manpower additional 40% for remainder of FY 80	5		-	
20. Further reduce staff manpower on OR routine actions by additional 5% (will defer work on NRC-required actions deemed of "Lesser safety significance")	4		-	
Totals	158		124	
Remainder of PMY in NRR	122		372	

Proposed Order of Reprogramming of NRR Manpower for Approved TMI-2 Actions

Brief Description of Reprogramming Action	Estimated Manpower Freed FY80 (7 mo) PMY	FY80 (7 mo) Cumu	FY81 (12 mo) PMY	FY81 (12 mo) Cumu
1. Defer work on Early Site Reviews thru FY 81	5	5	1	1
2. Eliminate Advanced Reactors work thru FY81	2	7	3	4
3. Defer standard plant reviews thru FY81	6	13	16	20
4. Substantially reduce NRR participation in standards development thru FY 81	3	16	4	24
5. Reduce NRR support to other offices to minimum level thru FY 81	5	21	12	36
6. Reduce work on revisions to Standard Review Plan and eliminate work on licensing improvements in FY 81	0	21	10	46
7. Defer work on CP applications thru FY 81, except those with complete SER	13	34	13	59
8. Substantially reduce effort on audit calculations thru FY 81	4	38	4	63
9. Defer work on non-critical topical reports thru FY 81	2	40	4	67
10. Defer all staff work on generic issues other than "Unresolved Safety Issues" thru FY 80, and reduce level of staff manpower on these issues to 40% of budget for FY 81	42	82	31	98
NTOL Reprogramming Line				
11. Reduce level of staff manpower on OL reviews by 50% for remainder of FY 80 and by 30% for FY 81 (will further delay issuances of scheduled reviews for FY 80 and 81)	18		26	
12. Reduce Post-CP work by one half	4		-	
13. Reduce level of staff manpower on OR routine actions by 10% for remainder of FY 80	9		-	
14. Stop all remaining work on OL reviews	17		-	
15. Reschedule USI as needed to reduce manpower by 40% for remainder of FY 80	5		-	
16. Reschedule SEP as needed to reduce manpower by 40% for remainder of FY 80	5		-	
17. Eliminate all Training for remainder of FY 80	4		-	
18. Reduce SEP by additional 40% (leaves only care-taker function) for remainder of FY 80	5		-	
19. Further reschedule USI to reduce manpower additional 40% for remainder of FY 80	5		-	
20. Further reduce staff manpower on OR routine actions by additional 5% (will defer work on NRC-required actions deemed of "Lesser safety significance")	4		-	
Totals	158		124	
Remainder of PMY in NRR	122		372	

ATTACHMENT E

X-n = Staff work in progress or was planned in FY80
CON = Working progress by contractor
RES
SD } = Work planned or in progress by other offices

Based on survey in late October 1979

RESULTS OF POINT ASSIGNMENTS

Point Total - 230

A-1 Water Hammer
A-3 Westinghouse Steam Generator Tube Integrity
A-7 Mark I - Long Term Program
A-9 ATWS
A-17 Systems Interactions
A-39 SRV Pool Dynamic Loads

Point Total - 220

A-2 Asymmetric Blowdown Loads
A-10 BWR Nozzle Cracking
A-24 Qualifications of Class IE Safety-Related Equipment
A-40 Seismic Design Criteria

Point Total - 210

A-11 Reactor Vessel Materials Toughness

Point Total - 200

A-36 Control of Heavy Loads Near Spent Fuel
A-4 CE Steam Generator Tube Integrity
A-5 B&W Steam Generator Tube Integrity
A-8 Mark II Program

Point Total - 190

A-42 Pipe Cracks in Boiling Water Reactors
A-44 Station Blackout

Point Total - 180

A-12 Steam Generator and Reactor Coolant Pump Support
B-6 Loads, Load Combinations, Stress Limits

ATTACHMENT F

Top 20

Point Total - 160

A-43 Containment Emergency Sump Reliability

Point Total - 150

A-20 Impacts of the Coal Fuel Cycle
A-33 NEPA Reviews of Accident Risks
Con B-41 Impacts on Fisheries

Point Total - 140

Con A-21 MSLB Inside Containment - Equipment Qualification
A-34 Instruments to Follow the Course of An Accident
A-35 Adequacy of Offsite Power Systems
B-26 Structural Integrity of Containment Penetrations
B-34 Occupational Radiation Exposure Reduction
B-56 Diesel Reliability
SD B-64 Decommissioning of Reactors

Available resources can be assigned at discretion of Division Director.

Point Total - 130

B-1 Environmental Technical Specifications
RES B-2 Forecasting Electricity Demand
SD B-46 Costs of Alternatives in Environmental Design

Point Total - 120

Con A-15 Primary Coolant System Decontamination
A-30 Adequacy of Safety-Related DC Power Supplies
A-37 Turbine Missiles
Con A-18 Pipe Rupture Design Criteria
B-48 BWR Control Rod Drive Mechanical Failures
RES B-37 Chemical Discharges to Receiving Water

Point Total - 110

Con A-14 Flaw Detection
A-25 Non-Safety Loads on Class IE Power Sources
RES A-38 Tornado Missiles
Con B-10 Behavior of BWR Mark III Containment
RES B-42 Socioeconomic Environmental Impacts

Point Total - 100

Con A-13 Snubbers
A-19 Digital Computer Protection Systems
A-28 Increase in Spent Fuel Pool Storage Capacity
A-29 Nuclear Power Plant Design for the Reduction of Vulnerability to Industrial Sabotage
Con B-5 Ductility of Two-Way Slabs and Shells and Buckling Behavior of Steel Containments
Con B-15 CONTEMPT Computer Code Maintenance
Con B-20 Standard Problem Analysis
Res B-25 Piping Benchmark Problems
Res B-40 Effects of Power Plant Entrainment on Plankton
C-10 Effective Operation of Containment Sprays in a LOCA

Expenditure of manpower halted as soon as practicable unless Director, NRR approves further work on these tasks.

Point Total - 90

A-27 Reload Application Guide
B-3 Event Categorization
B-22 LWR Fuel
B-59 N-1 Loop Operation in BWRs and PWRs
Con B-54 Ice Condenser Containment
B-66 Control Room Infiltration Measurements
C-1 Long Term Integrity of Seals on Instruments Inside Containment

Point Total - 80

A-22 PWR Main Steamline Break
Res B-28 Radionuclide/Sediment Transport Program
B-8 Locking Out of ECCS Power Operated Valves
B-55 Improved Reliability of Target lock Safety Relief Valve
B-53 Load Break Switch
B-61 Analytically Derived Allowable ECCS Equipment Outage Periods
B-71 Incident Response

Point Total - 70

A-16 Steam Effects on BWR Core Spray Distribution
A-32 Missile Effects
B-4 ECCS Reliability
B-9 Electrical Cable Penetrations of Containment
Con B-11 Subcompartment Standard Problems
B-12 Containment Cooling Requirements (Non-LOCA)
B-13 Marviken Test Data Evaluation
B-35 Confirmation of Appendix I Models

Point Total - 70 (continued)

con	B-47	Inservice Inspection of Supports Class 1, 2, 3 and MC Components
	B-39	Transmission Lines
con	B-44	Forecasts of Generating Costs of Coal and Nuclear Plants
	B-69	ECCS Leakage Ex-containment
	B-72	Health Effects and Life Shortening from Uranium and Coal Fuel Cycles
con	C-4	Statistical Methods for ECCS Analysis

Point Total - 60

con	B-19	Thermal Hydraulic Stability
	B-21	Core Physics
	B-49	Inservice Inspection Criteria and Corrosion Prevention Criteria for Containments
Complete	B-38	Reconnaissance Level Investigations
	B-43	Value of Aerial Photographs for Site Evaluation
	B-67	Effluent and Process Monitoring Information
	B-68	Pump Overspeed During a LOCA

Point Total - 50

con	B-17	Safety-Related Operator Actions
	B-24	Seismic Qualification of Electrical and Mechanical Equipment
	B-32	Ice Effects on Safety-Related Water Supplies
	B-50	Post-Operating Basis Earthquake
	B-51	Assessment of Inelastic Analysis Techniques for Equipment and Components
con	B-58	Passive Mechanical Failures
con	B-60	Loose Parts Monitoring
con	C-7	PWR System Piping
con	C-8	Main Steam Line Leakage Control Systems

Point Total - 40

	B-14	Study of Hydrogen Mixing Capability in Containment Post-LOCA
	B-27	Implementation and Use of Subsection NF
	B-62	Re-examination of Technical Bases for Establishing Technical Specifications Limits
	C-5	Decay Heat Update
	C-6	LOCA Heat Sources
	C-9	RHR Heat Exchanger Tube Failures
RES	C-16	Assessment of Agricultural Land in Relation to Power Plant Siting and Cooling System Selection

Point Total - 30

ccn B-7 Secondary Accident Consequence Modeling
B-31 Dam Failure Model
B-33 Dose Assessment Methodology
cmpltc B-36 Develop, Design Testing and Maintenance Criteria for Atmosphere Cleanup System
RES B-29 Effectiveness of Ultimate Heat Sinks
B-65 Iodine Spiking
B-73 Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel
C-2 Study of Containment Depressurization by Inadvertent Spray Operation
C-17 Interim Acceptance Criteria for Solidification Agents for Radioactive Solid Wastes

Point Total - 20

C-3 Insulation Usage Within Containment (Blocking of Vent Paths in Subcompartments)
ccn C-11 Assessment of Failure and Reliability of Pumps and Valves
C-15 NUREG Report for Liquid Tank Failure Analysis

Point Total - 10

B-23 LMFBR Fuel
D-1 Advisability of Seismic Scram
B-70 Power Grid Frequency Degradation and Effect on Primary Coolant Pumps

Point Total - 0

B-70 Power Grid Frequency Degradation and Effect on Primary Coolant Pumps
C-14 Storm Surge Model for Coastal Sites

Complete

A-6 Mark I Short Term Program
A-23 Containment Leak Testing
A-26 Reactor Vessel Pressure Transient Protection
A-31 RHR Shutdown Requirements
B-30 Design Basis Floods and Probability
B-63 Isolation of Low Pressure Systems Connected to the Reactor Coolant Pressure Boundary
D-3 Control Rod Drop Accident

Combined With Other Tasks

B-45	Need for Power-Energy Conservation (Included in B-2)
B-52	Fuel Assembly Seismic and LOCA Responses (Included in A-2)
B-57	Station Blackout (Included in A-44)
B-18	Vortex Suppression Requirements for Containment Sumps (Included in A-43)
B-16	Protection Against Postulated Piping Failures in Fluid Systems Outside Containment (Included in A-18)
C-12	Primary System Vibration (Included in B-73)
C-13	Non-Random Failures (Included in A-9, A-17, A-30, A-35, A-44 and B-56)
D-2	Emergency Core Cooling System Capability for Future Plants (Included in RES Improved Safety Research Program)