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

APR 1 2 1979

Directors of Offices

The attached information on Three Mile Island is provided for your information because of current interest in the matter.

(Signed) T. A. Rehm

T. A. Rehm, A/EDO

1. Chairman's Testimony of 4/10/79
2. Chronology of NRC Actions (preliminary)
3. Description of Events at TMI-2 Facility Accident (preliminary)
4. Generic Considerations of TMI-2 Incident
5. Summary of Initial Response & Radiological Surveys
6. Summary of Radioactive Liquid Release Situation
7. Preliminary Evaluation of Health Effects of the TMI Incident
8. Resident Inspection Program
9. Population Dose Estimates - Update of April 1 NRC/HEW/EPA study

TESTIMONY OF CHAIRMAN HENDRIE
BEFORE THE
SUBCOMMITTEE ON NUCLEAR REGULATION OF THE
SENATE COMMITTEE ON ENVIRONMENT AND PUBLIC WORKS
TUESDAY, APRIL 10, 1979

Mr. Chairman, as you and I agreed in arranging this hearing, we are here to present a preliminary account of what happened at Three Mile Island, our response to the accident, and some initial conclusions in order to put these matters on record at an early time. I am very grateful for and appreciative of your understanding that we could not and should not divert any appreciable staff resources from the operations at Three Mile Island or from the support groups at Bethesda to prepare a more extensive record for this hearing. There will, of course, be more detailed investigations, and we will look forward to future hearings in which a further accounting can be made.

First, by way of general comment, I want to emphasize my great personal concern, which I know is shared by my colleagues on the Commission and all members of our staff, that this accident at Three Mile Island Unit 2 has happened. All of us are profoundly ~~thankful~~ grateful, as I am sure you and all of our citizens are, that the radiation exposure levels to the public in the area of Three Mile Island have been low. However, we do not take much comfort from the low radiation exposures experienced to date since it is clear that the potential for substantially larger exposures of the public existed during the course of this accident.

ENCLOSURE 1

The efforts of a very large number of people in the licensee's organization, the ~~State~~ Commonwealth of Pennsylvania, the NRC, other Federal and State agencies, and industry groups are being applied to make sure that we can collectively maintain and improve the conditions that have prevailed thus far with regard to the offsite public and the workers at the site. The rapid marshalling of very extensive resources from a wide variety of sources has been a heartening experience and I believe the record should include an expression of our appreciation to the hundreds of individuals and organizations that have responded.

It is my view, and I ~~am sure~~ believe it is yours as well, that we cannot have an acceptable nuclear power program in this country if there is any appreciable risk of events of the Three Mile Island kind occurring at nuclear power plants. The Nuclear Regulatory Commission must promptly carry out a searching review and evaluation of our own policies and procedures, in addition to our investigation of what has taken place at the Three Mile Island facility. We must find out where our inspection and enforcement of safety-related operating requirements, our design standards, and our reviews of possible transient and accident situations have somehow been inadequate to prevent the Three Mile Island accident. We already have put those elements of the staff that are not immediately involved in dealing with the situation at Three Mile Island to work on this essential and major effort.

The Nuclear Regulatory Commission's investigation of the Three Mile Island accident is actively underway. Of course, our most immediate concern has been dominated by the operational considerations of limiting further releases and returning the plant to a safe and secure shutdown condition. Consistent with our efforts in that direction and to the extent we could do so without interfering with the recovery operation, our investigators have been at work gathering information. The NRC investigation team is now at the Three Mile Island site in the field phase of its work, following several days of review of plant records and charts furnished to our Operations Center in Bethesda.

As you are aware, Mr. Chairman, the NRC deliberated very carefully concerning the possibility of serious generic concerns that might require shutdown of the other nuclear power plants designed by Babcock and Wilcox, the nuclear steam system supplier for the Three Mile Island plant. The results of our review in this area led us to the conclusion that although shutdowns were not warranted at other B&W plants, it was important to require these other utilities to make changes in operating procedures and to carry out supplementary training of their operators.

These required actions were taken promptly and are being followed up by increased inspection activities.

First, an NRC Inspection and Enforcement Bulletin was transmitted on April 1, directing licensees operating B&W reactors to perform a series of specific reviews and actions. The licensees must respond by tomorrow. Our onsite inspectors will monitor compliance with this Bulletin. The NRC Headquarters staff will review the responses promptly and act upon them accordingly. This Bulletin was supplemented by another on April 5 that provided additional specific operating instructions based on our augmented understanding of the events at Three Mile Island. (Copies of these Bulletins are attached for the record.)

Second, an inspector was assigned full-time at each operating plant having a B&W reactor by April 2. In addition to assuring that the Bulletins are fully understood and followed, the full-time inspector -- with additional assistance from the NRC Regional Offices -- will assure that some inspection activities will occur during each shift and that each shift will have a clear understanding of the guidance. Regional and Headquarters staffs are staying in close touch with the onsite NRC inspectors to be sure that our instructions to the licensees are being carried out. Furthermore, the Commission has sent a telegram to each of the licensees with B&W reactors to underscore the seriousness with which the Commission views this situation. (A copy is attached for the record.)

Internally, an NRC Task Force on Generic Review of Feedwater Transients in B&W Reactors has been formed to examine the reactor and plant systems at these plants that provided protection against feedwater transients. The Task Force report is expected to be completed before the end of this month. We will carefully review that information and take whatever further action may be appropriate.

Pending completion of our review of the responses to the April 1 Bulletin and in view of the actions which the NRC has taken and our current understanding that operator errors played an important role in the accident, we believe there is reasonable assurance that the plants can continue to operate without danger to the public health and safety. I would note that Although Commissioner Bradford agrees fully that the aforementioned specific procedures are prudent and provide a considerably enhanced level of assurance, he would await the report of this Task Force due end of April before sharing fully in this conclusion.

Before describing the situation at Three Mile Island, I should like to comment on President Carter's role. From the outset we have kept the President fully informed concerning the accident. He has been most helpful in providing us with his personal views and suggestions and in making available to the NRC a communications network. Moreover, his visit to the site on April 1 helped alleviate the growing anxiety of the people in the area.

In his recent energy address, the President expressed concern about the accident and directed that an independent Presidential Commission be established to investigate the causes of the accident and to make recommendations on how the safety of nuclear power plants can be improved. The Nuclear Regulatory Commission will, of course, cooperate fully with the Presidential Commission in every way.

Initial Events

At about 4:00a.m. on March 28, 1979, an event occurred which lead to the accident at the Three Mile Island Unit No. 2 facility. The sequence of events included a loss of feedwater flow, the sticking open of a primary relief valve, the premature turning off of the emergency core cooling system, and later the turning off of all reactor coolant pumps. Shortly before 7:00a.m. a site emergency was declared by the utility as it became apparent that the reactor fuel had been damaged.

I would now like to have Mr. Darrell Eisenhut go through the detailed sequence of events, as we understand them now, in the first hours of the accident. Then I will resume my statement.

resumes 10:22

At about 7:00 a.m. on March 28th, the licensee notified the State of Pennsylvania. At 7:45 a.m., the licensee reached the NRC Region I office, and within ½ hour the NRC Incident Response Center in Bethesda was operating. The White House was alerted at 9:15 a.m., and by 10:05 a.m. the first NRC Response Team was on the TMI site, a group of five inspectors from our Region I office. At 10:15 a.m. the Commission was briefed by the Director of the Office of Inspection and Enforcement. NRC had 11 people and a mobile laboratory van from the Region I Office on site by evening; radiological assistance teams from Brookhaven Laboratory were monitoring the site vicinity from mid-afternoon on. An Aerial Monitoring Survey aircraft also was at the site by mid-afternoon and making tracking flights. NRC strength at the site increased to 29 on Thursday and 83 on Friday. (A brief chronology of these response actions is attached for the record.)

From Wednesday on, there were continuing releases of radioactive gas evolving from the reactor cooling water which had been pumped from the containment building to the auxiliary building. Despite efforts to halt these releases, they increased on Friday morning. This situation led to a decision by Governor Thornburgh -- on the Commission's recommendation -- to advise preschool children and pregnant women to leave the 5-mile zone nearest the reactor. The releases were greatly reduced on Friday afternoon when some of the water in the auxiliary building was pumped to storage tanks emptied for that purpose. NRC increased the level of its onsite team, and Harold Denton, the Director of Nuclear Reactor Regulation, was placed in charge at the site. Communications and transportation support were supplied at White House direction. Other Federal agencies provided additional assistance.

NRC received the full cooperation of Pennsylvania State government officials as well as other Federal agencies. Nowhere was this cooperation more apparent than in the vitally important area of radiological monitoring. For instance, the Department of Energy's Aerial Monitoring Survey airplane was on station over the site within a few hours. Personnel from the Pennsylvania State Bureau of Radiological Health were making measurements from Thursday afternoon onward. Radiological monitoring efforts were also carried out by HEW and EPA teams.

resumes 6935

On Friday, it was recognized that the early core damage was more extensive than had been thought had formed a substantial quantity of hydrogen gas, some of which was thought to have collected in the reactor pressure vessel above the core. One concern was that if the reactor pressure was decreased, the hydrogen bubble would grow and might interfere with the flow of cooling water through the core. Another was that oxygen generated by radiolysis of water might accumulate -- with the potential for forming a flammable mixture. The Commission considered recommending additional evacuation measures beyond those previously recommended to Governor Thornburgh -- which I indicated in my press conference on Saturday afternoon -- but decided that increased protective action was not necessary. I should note that at one time or another from Friday morning through Sunday, I would guess, members of the senior staff and various Commissioners proposed either stronger advisories or actual evacuation of people within a two or five mile circle around Three Mile Island. However, the estimates from ~~NRC~~ Mr. Denton & his staff at the site remained considerably more optimistic than those which could be based on the more limited information available ~~to us in Washington or Bethesda~~ up here. Consequently, stronger recommendations were not made to Governor Thornburg. The precise timing and content of the various proposed Commission recommendations will certainly be made public as soon as all of the relevant tapes and notes can be reviewed and transcribed. The information will then be promptly furnished to the Congress.

resume 1048

By Sunday, April 1, it appeared that efforts to understand and manage the hydrogen bubble were meeting with some success. Over the following few days, the bubble was reduced to negligible size by degassing with the pressurizer spray and the letdown flow and by gas dissolving in the reactor coolant water. Further analysis had also indicated that little, if any, oxygen could be evolved in the hydrogen-rich conditions of the reactor system. Thus, the period of immediate crisis passed and the reactor cooldown process could proceed without the hydrogen bubble problem.

The entire recovery operation is being conducted in a way that minimizes the release of radioactivity from the site and that requires very careful consideration and concurrence by the on-site NRC team in every significant action taken. With the reactor in a stable cooling status since early in the accident, we have been very cautious about moving too rapidly in the recovery operations lest we encounter situations that have not been carefully thought out. I am convinced that this is the correct approach, even though, from the public's point of view, it may seem to prolong the emergency.

1054

Let me now give you a brief status report on the current situation at Three Mile Island. I was at the site again last Sunday and have been in close contact with Harold Denton and his team at the site throughout the accident.

Conclusion

The staff has identified six main factors that caused and increased the severity of the accident. These are as follows:

1. At the time of the initiating event, loss of feedwater, both of the auxiliary feedwater trains were valved out of service. This was a violation of the plant Technical Specifications.

2. The pressurizer electromatic relief valve, which opened during the initial pressure surge, failed to close when the pressure decreased below the actuation level. This failure was not recognized and the relief line closed for some time.

3. Following rapid depressurization of the pressurizer, the pressurizer level indication may have lead to erroneous inferences of high level in the reactor coolant system. The pressurizer level indication apparently led the operators to prematurely terminate high pressure injection flow, even though substantial voids existed in the reactor coolant system.

4. Because the containment does not isolate on high pressure injection (HPI) initiation, the highly radioactive water from the relief valve discharge was pumped out of the containment by the automatic initiation of a transfer pump. This water entered the radioactive waste treatment system in the auxiliary building where some of it overflowed to the floor. Outgassing from this water and discharge through the auxiliary building ventilation system and filters was the principal source of the offsite release of radioactive noble gases.

5. Subsequently, the high pressure injection system was only intermittently operated to control primary coolant inventory losses through the electromatic relief valve, apparently based on pressurizer level indication. Due to the presence of steam and/or noncondensible voids elsewhere in the reactor coolant system, this led to a further reduction in primary coolant inventory.

6. Tripping of reactor coolant pumps during the course of the transient, presumably to protect against pump damage due to pump vibration, led to fuel damage since voids in the reactor coolant system prevented natural circulation.

As I remarked at the beginning, I am deeply concerned about the accident at Three Mile Island and its implications. Even though, as far as we can determine at this time, no one was exposed to dangerous radiation levels, the accident was indeed serious. The lives of the people in the neighboring communities were disrupted and many individuals experienced personal hardships. And, of course, the economic consequences of the accident are substantial. We cannot tolerate accidents of this kind and we must take whatever steps are necessary to prevent them.

We have a lot to learn from the Three Mile Island accident. I would like to emphasize here that I'm not just thinking of improved hardware or procedures, or other technical fixes. I'm thinking of the regulatory framework itself. I do not see the Three Mile Island accident and its implications as matters that can be addressed exclusively on a technical plane.

For example, the total response of all parties involved reveals that a critical reexamination of the relationship among licensees, regulators and support groups during crisis management is needed. I do not mean to convey any impression of conflict among these groups arising out of the Three Mile Island accident. Indeed, cooperation has been very good. However, during these last 2 weeks we in NRC have been heavily involved in operational matters that we normally expect to be primarily the licensee's responsibility, subject to our review and approval. Our involvement in these matters has been appropriate under the circumstances, but we believe that we now should look more carefully at our preplanned role in crises. It is possible that legislative action may be needed in this area.

We have been sorely taxed to assemble NRC personnel from our other Regional Offices and our licensing offices to provide onsite NRC staff support. We may find that our budget planning has inadequately considered the impact of short-term heavy demands such as the Three Mile Island accident.

It is already clear to me that among the likely areas where improvements should be made are first, a substantial upgrading of reactor operator training with respect to plant transients and off-normal conditions. Second, we should put more emphasis in our staff reviews and safety regulations on safety measures to deal with plant transients originating in the secondary or steam-producing side of nuclear power plants. Third, we must carefully reexamine the sensitivity of all plant designs to these transient situations and we must reexamine the automatic safety systems that deal with them. Finally, we must make certain that our licensing review and our inspection procedures are as capable as we can make them of preventing the mistakes that were made at Three Mile Island.

ATTACHMENT 1

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

April 1, 1979

MEMORANDUM FOR: B. H. Grier, Director, Region I
J. P. O'Reilly, Director, Region II
J. G. Keppler, Director, Region III
K. V. Seyfrit, Director, Region IV
R. H. Engelken, Director, Region V

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, OIE

SUBJECT: IE BULLETIN 79-05, NUCLEAR INCIDENT AT THREE MILE ISLAND

The subject IE Bulletin should be dispatched for action by April 2, 1979, to all B&W power reactor facilities with an operating license.

Subject bulletin and enclosures should also be dispatched for information to all other power reactor facilities with an operating license and to all power reactor construction permit holders.

The text of the Bulletin, Enclosure 2 and draft letters to the licensee are enclosed for this purpose. Enclosure 1 which consists of the referenced Preliminary Notifications, should be added by the regional office. The letters to the licensee make the commitment to forward the continuing Preliminary Notifications of the incident. These should be forwarded as they are received.

Norman C. Moseley

Norman C. Moseley, Director
Division of Reactor Operations
Inspection
Office of Inspection and Enforcement

Enclosures:
1. Draft Transmittal Letters to B&W Licensees
2. Draft Transmittal Letter to all other power facilities
3. IE Bulletin No. 79-05
4. Enclosure 2, to the Bulletin

CONTACT: D.C. Kirkpatrick, IE
49-28180

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

April 1, 1979

IE Bulletin No. 79-05

NUCLEAR INCIDENT AT THREE MILE ISLAND

Description of Circumstances:

On March 28, 1979 the Three Mile Island Nuclear Power Plant, Unit 2 experienced core damage which resulted from a series of events which were initiated by a loss of feedwater transient. Several aspects of the incident may have general applicability in addition to apparent generic applicability at operating Babcock and Wilcox reactors. This bulletin is provided to inform you of the nuclear incident and to request certain actions.

Actions To Be Taken By Licensees

(Although the specific causes have not been determined for individual sequences in the Three Mile Island event, some of the following may have contributed.)

For all Babcock and Wilcox pressurized water reactor facilities with an operating license:

1. Review the description (Enclosure 1) of the initiating events and subsequent course of the incident. Also review the evaluation by the NRC staff of a postulated severe feedwater transient related to Babcock and Wilcox PWRs as described in Enclosure 2.

   These reviews should be directed at assessing the adequacy of your reactor systems to safely sustain cooldown transients such as these.

2. Review any transients of a similar nature which have occurred at your facility and determine whether any significant deviations from expected performance occurred. If any significant deviations are found, provide the details and an analysis of the significance and any corrective actions taken. This material may be identified by reference if previously submitted to the NRC.

3. Review the actions required by your operating procedures for coping with transients. The items that should be addressed include:

a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability.

b. Operator action required to prevent the formation of such voids.

c. Operator action required to ensure continued core cooling in the event that such voids are formed.

4. Review the actions requested by the operating procedures and the training instructions to assure that operators do not override automatic actions of engineered safety features without sufficient cause for doing so.

5. Review all safety related valve positions and positioning requirements to assure that engineered safety features and related equipment such as the auxiliary feedwater system, can perform their intended functions. Also review related procedures, such as those for maintenance and testing, to assure that such valves are returned to their correct positions following necessary manipulations.

6. Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the containment to assure that undesired pumping of radioactive liquids and gases will not occur inadvertently.

   In particular assure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

   a. Whether interlocks exist to prevent transfer when high radiation indication exists and,

   b. Whether such systems are isolated by the containment isolation signal.

7. Review your prompt reporting procedures for NRC notification to assure very early notification of serious events.

The detailed results of these reviews shall be submitted within ten (10) days of the receipt of this Bulletin.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Construction Inspection, Washington, D.C. 20555.

For all other operating reactor or reactors under construction, this Bulletin is for information purposes and no report is requested.

Approved by GAO, B180225 (R0072); clearance expires 7-31-80. Approval was given under a blanket clearance specifically for identified generic problems.

(Draft letter to B&W power reactor facilities with an operating license)

IE Bulletin No. 79-05

Addressee:

Enclosed is IE Bulletin No. 79-05, which requires action by you with regard to your power reactor facility(ies) with an operating license or a construction permit.

Should you have any questions regarding this Bulletin or the actions required by you, please contact this office.

Sincerely,

Signature
(Regional Director)

Enclosures:
1. IE Bulletin No. 79-05
and Enclosures
2. List of IE Bulletins
issued in the past
12 months

(Draft letter to all power reactor facilities with an operating license or a construction permit)

IE Bulletin No. 79-05

Addressee:

The enclosed Bulletin 79-05 is forwarded to you for information. No written response is required. If you desire additional information regarding this matter, please contact this office. The Preliminary Notification of the subject incident (Enclosure 1) will continue to be issued periodically. These will be forwarded to you as they are issued.

Sincerely,

Signature
(Regional Director)

Enclosures:
1. IE Bulletin No. 79-05
   and Enclosures
2. List of IE Bulletins
   issued in the past
   12 months

ENCLOSURE 2

A loss of offsite power occurred at Davis-Besse on November 29, 1977, which resulted in shrinkage of the primary coolant volume to the degree that pressurizer level indication was lost. A recommendation to convey this information to certain hearing boards resulted in the attached discussion and evaluation of the event. This discussion includes a review of a loss of feedwater safety analysis assuming forced flow, which predicts dispersed primary system voiding, but no loss of core cooling. During the Three Mile Island event, however, the forced flow appears to have been terminated during the transient.

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
78-05	Malfunctioning of Circuit Breaker Auxiliary Contact Mechanism-General Model CR105X	4/14/78	All Power Reactor Facilities with an OL or CP
78-06	Defective Cutler-Hammer, Type M Relays With DC Coils	5/31/78	All Power Reactor Facilities with an OL or CP
78-07	Protection afforded by Air-Line Respirators and Supplied-Air Hoods	6/12/78	All Power Reactor Facilities with an OL, all class E and F Research Reactors with an OL, all Fuel Cycle Facilities with an OL, and all Priority 1 Material Licensees
78-08	Radiation Levels from Fuel Element Transfer Tubes	6/12/78	All Power and Research Reactor Facilities with a Fuel Element transfer tube and an OL.
78-09	BWR Drywell Leakage Paths Associated with Inadequate Drywell Closures	6/14/79	All BWR Power Reactor Facilities with an OL or CP
78-10	Bergen-Paterson Hydraulic Shock Suppressor Accumulator Spring Coils	6/27/78	All BWR Power Reactor Facilities with an OL or CP

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
78-11	Examination of Mark I Containment Torus Welds	7/21/78	BWR Power Reactor Facilities for action: Peach Bottom 2 and 3, Quad. Cities 1 and 2, Hatch 1, Monticello and Vermont Yankee
78-12	Atypical Weld Material in Reactor Pressure Vessel Welds	9/29/78	All Power Reactor Facilities with an OL or CP
78-12A	Atypical Weld Material in Reactor Pressure Vessel Welds	11/24/78	All Power Reactor Facilities with an OL or CP
78-12B	Atypical Weld Material in Reactor Pressure Vessel Welds	3/19/79	All Power Reactor Facilities with an OL or CP
78-13	Failures In Source Heads of Kay-Ray, Inc., Gauges Models 7050, 7050B, 7051, 7051B, 7060, 7060B, 7061 and 7061B	10/27/78	All general and specific licensees with the subject Kay-Ray, Inc. gauges
78-14	Deterioration of Buna-N Components In ASCO Solenoids	12/19/78	All GE BWR facilities with an OL or CP
79-01	Environmental Qualification of Class IE Equipment	2/8/79	All Power Reactor Facilities with an OL or CP

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-02	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	3/2/79	All Power Reactor Facilities with an OL or CP
79-03	Longitudinal Weld Defects In ASME SA-312 Type 304 Stainless Steel Pipe Spools Manufactured By Youngstown Welding and Engineering Co.	3/12/79	All Power Reactor Facilities with an OL or CP
79-04	Incorrect Weights for Swing Check Valves Manufactured by Velan Engineering Corporation	3/30/79	All Power Reactor Facilities with an OL or CP

IE Bulletin No. 79-05 April 1, 1979

INCORRECT WEIGHTS FOR SWING CHECK VALVES MANUFACTURED BY VELAN ENGINEERING CORPORATION

Distribution:

Technical Assistant, OCM (Chairman's Office)	H-1149
George Eysymontt, Technical Assistant, OCM (Gilinsky)	H1149
John C. Guibert, Technical Assistant, OCM (Kennedy)	H1149
Hugh L. Thompson, Technical Assistant, OCM (Bradford)	H1149
George D. Sauter, Technical Assistant, OCM (Ahearne)	H1149
SECY	H-1149
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L. Bickwit, General Counsel	H-1047
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R. S. Boyd, Dir., DPM:NRR	Phil-268
D. F. Ross, Deputy Director, DPM:NRR	Phil-278
V. Stello, Dir., DOR:NRR	Phil-542
J. R. Miller, DOR:NRR	Phil-216
D. G. Eisenhut, Dep. Dir., DOR:NRR	Phil-266
R. H. Vollmer, Asst. Dir., SP:DOR:NRR	P-514
G. C. Lainas, Chief, PSB:DOR:NRR	Phil-416
B. K. Grimes, Asst. Dir., E/P:DOR:NRR	Phil-370
R. J. Mattson, Dir., DSS:NRR	Phil-202
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E. M. Howard, SI:IE	
L. I. Cobb, XOMA:IE	
D.C. Kirkpatrick, IE contact	

EXCERPT FROM MEMORANDUM ENTITLED "CONVEYING NEW INFORMATION TO LICENSING BOARDS - DAVIS-BESSE UNITS 2 & 3 AND MIDLAND UNITS 1 & 2", DATED JANUARY 8, 1979, FROM J.S. CRESWELL TO J.F. STREETER.

3. Inspection and Enforcement Report 50-346/78-06 documented that pressurizer level had gone offscale for approximately five minutes during the November 29, 1977 loss of offsite power event. There are some indications that other B&W plants may have problems maintaining pressurizer level indications during transients. In addition, under certain conditions such as loss of feedwater at 100% power with the reactor coolant pumps running the pressurizer may void completely. A special analysis has been performed concerning this event. This analysis is attached as Enclosure 1. Because of pressurizer level maintenance problems the sizing of the pressurizer may require further review.

Also noted during the event was the fact that Tcold went offscale (less than 520°F). In addition, it was noted that the makeup flow monitoring is limited to less than 160 gpm and that makeup flow may be substantially greater than this value. This information should be examined in light of the requirements of GDC 13.

DISCUSSION AND EVALUATION

The event at Davis Besse which resulted in loss of pressurizer level indication has been reviewed by NRR and the conclusion was reached that no unreviewed safety question existed.

The pressurizer, together with the reactor coolant makeup system, is designed to maintain the primary system pressure and water level within their operational limits only during normal operating conditions. Cooldown transients, such as loss of offsite power and loss of feedwater, sometimes result in primary pressure and volume changes that are beyond the ability of this system to control. The analyses of and experience with such transients show, however, that they can be sustained without compromising the safety of the reactor. The principal concern caused by such transients is that they might cause voiding in the primary coolant system that would lead to loss of ability to adequately cool the reactor core. The safety evaluation of the loss of offsite power transient shows that, though level indication is lost, some water remains in the pressurizer and the pressure does not decrease below about 1600 psi. In order for voiding to occur, the pressure must decrease below the saturation pressure corresponding to the system temperature. 1600 psi is the saturation pressure corresponding to 605°F, which is also the maximum allowable core outlet temperature. Voiding in the primary system (excepting the pressurizer) is precluded in this case, since pressure does not decrease to saturation.

The safety analysis for more severe cooldown transients, such as the loss of feedwater event, indicates that the water volume could decrease to less than the system volume exclusive of the pressurizer. During such an event, the emptying of the pressurizer would be followed by a pressure reduction below the saturation point and the formation of small voids throughout much of the primary system. This would not result in the loss of core cooling because the voids would be dispersed over a large volume and forced flow would prevent them from coalescing sufficiently to prevent core cooling. The high pressure coolant injection pumps are started automatically when the primary pressure decreases below 1600 psi. Therefore, any pressure reduction which is sufficient to allow voiding will also result in water injection which will rapidly restore the primary water to normal levels.

For these reasons, we believe that the inability of the pressurizer and normal coolant makeup system to control some transients does not provide a basis for requiring more capacity in these systems.

General Design Criterion 13 of Appendix A to 10 CFR 50 requires instrumentation to monitor variables over their anticipated ranges for "anticipated operational occurrences". Such occurrences are specifically defined to include loss of all offsite power. The fact that T cold goes off scale at 520°F is not considered to be a deviation from this requirement because this indicator is backed up by wide range temperature indication that extends to a low limit of 50°F. Neither do we consider the makeup flow monitoring to deviate since the amount of makeup flow in excess of 160 gpm does not appear to be a significant factor in the course of these occurrences.

The loss of pressurizer water level indication could be considered to deviate from GDC 13, because this level indication provides the principal means of determining the primary coolant inventory. However, provision of a level indication that would cover all anticipated occurrences may not be practical. As discussed above, the loss of feedwater event can lead to a momentary condition wherein no meaningful level exists, because the entire primary system contains a steam water mixture.

It should be noted that the introduction to Appendix A (last paragraph) recognizes that fulfillment of some of the criteria may not always be appropriate. This introduction also states that departures from the Criteria must be identified and justified. The discussion of GDC 13 in the Davis Besse FSAR lists the water level instrumentation, but does not mention the possibility of loss of water level indication during transients. This apparent omission in the safety analysis will be subjected to further review.

ATTACHMENT 2

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

April 5, 1979

MEMORANDUM FOR: B. H. Grier, Director, Region I
J. P. O'Reilly, Director, Region II
J. G. Keppler, Director, Region III
K. V. Seyfrit, Director, Region IV
R. H. Engelken, Director, Region V

FROM: Norman C. Moseley, Director, Division of Reactor Operations Inspection, OIE

SUBJECT: IE BULLETIN 79-05A, NUCLEAR INCIDENT AT THREE MILE ISLAND

The subject IE Bulletin should be dispatched for action by April 5, 1979, to all B&W power reactor facilities with an operating license. The Bulletin must be faxed to B&W facilities by April 5.

Subject bulletin and enclosures should also be dispatched for information to all other power reactor facilities with an operating license and to all power reactor facilities with a construction permit.

The text of the Bulletin, Enclosures thereto and draft letters ~~to the~~ licensee are enclosed for this purpose. The continuing Preliminary Notifications of the incident should continue to be forwarded as they are received in accordance with the transmittal memorandum for IE Bulletin 79-05.

for E L Jordan

Norman C. Moseley, Director
Division of Reactor Operations Inspection
Office of Inspection and Enforcement

Enclosures:
1. Draft Transmittal Letter to all B&W Operating Licensees
2. Draft Transmittal Letter to all other Operating Licensees and Construction Permit Holders.
3. IE Bulletin No. 79-05A (w/enclosures - 2)

CONTACT: D. C. Kirkpatrick, IE
49-28180

(Draft letter to B&W power reactor facilities with an operating license.)

IE Bulletin No. 79-05A

Addressee:

Enclosed is IE Bulletin No. 79-05A, which requires action by you with regard to your power reactor facility(ies) with an operating license.

Should you have any questions regarding this Bulletin or the actions required by you, please contact this office.

Sincerely,

Signature
(Regional Director)

Enclosure:
IE Bulletin No. 79-05A
with Enclosures

(Draft letter to all power reactor facilities with an operating license or a construction permit.)

IE Bulletin No. 79-05A

Addressee:

The enclosed Bulletin 79-05A is forwarded to you for information. No written response is required. If you desire additional information regarding this matter, please contact this office.

Sincerely,

Signature
(Regional Director)

Enclosure:
IE Bulletin No. 79-05A
with Enclosures

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, DC 20555

APRIL 5, 1979

IE Bulletin 79-05

NUCLEAR INCIDENT AT THREE MILE ISLAND - SUPPLEMENT

Description of Circumstances:

Preliminary information received by the NRC since issuance of IE Bulletin 79-05 on April 1, 1979 has identified six potential human, design and mechanical failures which resulted in the core damage and radiation releases at the Three Mile Island Unit 2 nuclear plant. The information and actions in this supplement clarify and extend the origina Bulletin and transmit a preliminary chronology of the TMI accident through the first 16 hours (Enclosure 1).

1. At the time of the initiating event, loss of feedwater, both of the auxiliary feedwater trains were valved out of service.

2. The pressurizer electromatic relief valve, which opened during the initial pressure surge, failed to close when the pressure decreased below the actuation level.

3. Following rapid depressurization of the pressurizer, the pressurize level indication may have lead to erroneous inferences of high level in the reactor coolant system. The pressurizer level indicat apparently led the operators to prematurely terminate high pressure injection flow, even though substantial voids existed in the reacto coolant system.

4. Because the containment does not isolate on high pressure injection (HPI) initiation, the highly radioactive water from the relief valve discharge was pumped out of the containment by the automatic initiation of a transfer pump. This water entered the radioactive waste treatment system in the auxiliary building where some of it overflowed to the floor. Outgassing from this water and discharge through the auxiliary building ventilation system and filters was the principal source of the offsite release of radioactive noble gases.

5. Subsequently, the high pressure injection system was intermittentl operated attempting to control primary coolant inventory losses through the electromatic relief valve, apparently based on pressurizer level indication. Due to the presence of steam and/or noncondensible voids elsewhere in the reactor coolant system, this led to a further reduction in primary coolant inventory.

6. Tripping of reactor coolant pumps during the course of the transient to protect against pump damage due to pump vibration, led to fuel damage since voids in the reactor coolant system prevented natural circulation.

Actions To Be Taken by Licensees:

For all Babcock and Wilcox pressurized water reactor facilities with an operating license (the actions specified below replace those specified in IE Bulletin 79-05):

1. (This item clarifies and expands upon item 1. of IE Bulletin 79-05.)

   In addition to the review of circumstances described in Enclosure 1 of IE Bulletin 79-05, review the enclosed preliminary chronology of the TMI-2 3/28/79 accident. This review should be directed toward understanding the sequence of events to ensure against such an accident at your facility(ies).

2. (This item clarifies and expands upon item 2. of IE Bulletin 79-05.)

   Review any transients similar to the Davis Besse event (Enclosure 2 of IE Bulletin 79-05) and any others which contain similar elements from the enclosed chronology (Enclosure 1) which have occurred at your facility(ies). If any significant deviations from expected performance are identified in your review, provide details and an analysis of the safety significance together with a description of any corrective actions taken. Reference may be made to previous information provided to the NRC, if appropriate, in responding to this item.

3. (This item clarifies item 3. of IE Bulletin 79-05.)

   Review the actions required by your operating procedures for coping with transients and accidents, with particular attention to:

   a. Recognition of the possibility of forming voids in the primary coolant system large enough to compromise the core cooling capability, especially natural circulation capability.

   b. Operator action required to prevent the formation of such voids.

   c. Operator action required to enhance core cooling in the event such voids are formed.

4. (This item clarifies and expands upon item 4. of IE Bulletin 79-05.)

Review the actions directed by the operating procedures and training instructions to ensure that:

a. Operators do not override automatic actions of engineered safety features.

b. Operating procedures currently, or are revised to, specify that if the high pressure injection (HPI) system has been automatically actuated because of low pressure condition, it must remain in operation until either:

(1) Both low pressure injection (LPI) pumps are in operation and flowing at a rate in excess of 1000 gpm each and the situation has been stable for 20 minutes, or

(2) The HPI system has been in operation for 20 minutes, and all hot and cold leg temperatures are at least 50 degrees below the saturation temperature for the existing RCS pressure. If 50 degree subcooling cannot be maintained after HPI cutoff, the HPI shall be reactivated.

c. Operating procedures currently, or are revised to, specify that in the event of HPI initiation, with reactor coolant pumps (RCP) operating, at least one RCP per loop shall remain operating.

d. Operators are provided additional information and instructions to not rely upon pressurizer level indication alone, but to also examine pressurizer pressure and other plant parameter indications in evaluating plant conditions, e.g., water inventory in the reactor primary system.

5. (This item revises item 5. of IE Bulletin 79-05.)

Verify that emergency feedwater valves are in the open position in accordance with item 8 below. Also, review all safety-related valve positions and positioning requirements to assure that valves are positioned (open or closed) in a manner to ensure the proper operation of engineered safety features. Also review related procedures, such as those for maintenance and testing, to ensure that such valves are returned to their correct positions following necessary manipulations.

6. Review the containment isolation initiation design and procedures, and prepare and implement all changes necessary to cause containment isolation of all lines whose isolation does not degrade core cooling capability upon automatic initiation of safety injection.

7. For manual valves or manually-operated motor-driven valves which could defeat or compromise the flow of auxiliary feedwater to the steam generators, prepare and implement procedures which:

   a. require that such valves be locked in their correct position; or

   b. require other similar positive position controls.

8. Prepare and implement immediately procedures which assure that two independent steam generator auxiliary feedwater flow paths, each with 100% flow capacity, are operable at any time when heat removal from the primary system is through the steam generators. When two independent 100% capacity flow paths are not available, the capacity shall be restored within 72 hours or the plant shall be placed in a cooling mode which does not rely on steam generators for cooling within the next 12 hours.

   When at least one 100% capacity flow path is not available, the reactor shall be made subcritical within one hour and the facility placed in a shutdown cooling mode which does not rely on steam generators for cooling within 12 hours or at the maximum safe shutdown rate.

9. (This item revises item 6 of IE Bulletin 79-05.)

   Review your operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of the primary containment to assure that undesired pumping of radioactive liquids and gases will not occur inadvertently.

   In particular, ensure that such an occurrence would not be caused by the resetting of engineered safety features instrumentation. List all such systems and indicate:

   a. Whether interlocks exist to prevent transfer when high radiation indication exists, and

   b. Whether such systems are isolated by the containment isolation signal.

10. Review and modify as necessary your maintenance and test procedures to ensure that they require:

    a. Verification, by inspection, of the operability of redundant safety-related systems prior to the removal of any safety-related system from service.

    b. Verification of the operability of all safety-related systems when they are returned to service following maintenance or testing.

    c. A means of notifying involved reactor operating personnel whenever a safety-related system is removed from and returned to service.

11. All operating and maintenance personnel should be made aware of the extreme seriousness and consequences of the simultaneous blocking of both auxiliary feedwater trains at the Three Mile Island Unit 2 plant and other actions taken during the early phases of the accident.

12. Review your prompt reporting procedures for NRC notification to assure very early notification of serious events.

For Babcock and Wilcox pressurized water reactor facilities with an operating license, respond to Items 1, 2, 3, 4.a and 5 by April 11, 1979. Since these items are substantially the same as those specified in IE Bulletin 79-05, the required date for response has not been changed. Respond to Items 4.b through 4.d, and 6 through 12 by April 16, 1979.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, DC 20555.

For all other reactors with an operating license or construction permit, this Bulletin is for information purposes and no written response is required.

Approved by GAO, B 180225 (R0072); clearance expires 7-31-80. Approval was given under a blanket clearance specifically for identified generic problems.

Enclosures:

1. Preliminary Chronology of TMI-2 3/38/79 Accident Until Core Cooling Restored.

2. List of IE Bulletins issued in last 12 months.

PRELIMINARY

CHRONOLOGY OF TMI-2 3/28/79 ACCIDENT UNTIL CORE COOLING RESTORED

TIME (Approximate)	EVENT
about 4 AM (t = 0)	Loss of Condensate Pump Loss of Feedwater Turbine Trip
t = 3-6 sec.	Electromatic relief valve opens (2255 psi) to relieve pressure in RCS
t = 9-12 sec.	Reactor trip on high RCS pressure (2355 psi)
t = 12-15 sec.	RCS pressure decays to 2205 psi (relief valve should have closed)
t = 15 sec.	RCS hot leg temperature peaks at 611 degrees F, 2147 psi (450 psi over saturation)
t = 30 sec.	All three auxiliary feedwater pumps running at pressure (Pumps 2A and 2B started at turbine trip). No flow was injected since discharge valves were closed.
t = 1 min.	Pressurizer level indication begins to rise rapidly
t = 1 min.	Steam Generators A and B secondary level very low - drying out over next couple of minutes.
t = 2 min.	ECCS initiation (HPI) at 1600 psi
t = 4 - 11 min.	Pressurizer level off scale - high - one HPI pump manually tripped at about 4 min. 30 sec. Second pump tripped at about 10 min. 30 sec.
t = 6 min.	RCS flashes as pressure bottoms out at 1350 psig (Hot leg temperature of 584 degrees F)
t = 7 min., 30 sec.	Reactor building sump pump came on.

TIME	EVENT
t = 8 min.	Auxiliary feedwater flow is initiated by opening closed valves
t = 8 min. 18 sec.	Steam Generator B pressure reached minimum
t = 8 min. 21 sec.	Steam Generator A pressure starts to recover
t = 11 min.	Pressurizer level indication comes back on scale and decreases
t = 11-12 min.	Makeup Pump (ECCS HPI flow) restarted by operators
t = 15 min.	RC Drain/Quench Tank rupture disk blows at 190 psig (setpoint 200 psig) due to continued discharge of electromatic relief valve
t = 20 - 60 min.	System parameters stabilized in saturated condition at about 1015 psig and about 550 degrees F.
t = 1 hour, 15 min.	Operator trips RC pumps in Loop B
t = 1 hour, 40 min.	Operator trips RC pumps in Loop A
t = 1-3/4 - 2 hours	CORE BEGINS HEAT UP TRANSIENT - Hot leg temperature begins to rise to 620 degrees F (off scale within 14 minutes) and cold leg temperature drops to 150 degrees F. (HPI water)
t = 2.3 hour	Electromatic relief valve isolated by operator after S.G.-B isolated to prevent leakage
t = 3 hours	RCS pressure increases to 2150 psi and electromatic relief valve opened
t = 3.25 hours	RC drain tank pressure spike of 5 psig
t = 3.8 hours	RC drain tank pressure spike of 11 psi - RCS pressure 1750; containment pressure increases from 1 to 3 psig
t = 5 hours	Peak containment pressure of 4.5 psig
t = 5 - 6 hours	RCS pressure increased from 1250 psi to to 2100 psi

TIME	EVENT
t = 7.5 hours	Operator opens electromatic relief valve to depressurize RCS to attempt initiation of RHR at 400 psi
t = 8 - 9 hours	RCS pressure decreases to about 500 psi Core Flood Tanks partially discharge
t = 10 hour	28 psig containment pressure spike, containment sprays initiated and stopped after 500 gal. of NaOH injected (about 2 minutes of operation)
t = 13.5 hours	Electromatic relief valve closed to repressurize RCS, collapse voids, and start RC pump
t = 13.5 - 16 hours	RCS pressure increased from 650 psi to 2300 psi
t = 16 hours	RC pump in Loop A started, hot leg temperature decreases to 560 degrees F, and cold leg temperature increases to 400 degrees F. indicating flow through steam generator
Thereafter	S/G "A" steaming to condensor Condensor vacuum re-established RCS cooled to about 280 degrees F., 1000 psi
Now (4/4)	High radiation in containment All core thermocouples less than 460 degrees F. Using pressurizer vent valve with small makeup flow Slow cooldown RB pressure negative

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
78-05	Malfunctioning of Circuit Breaker Auxiliary Contact Mechanism-General Model CR105X	4/14/78	All Power Reactor Facilities with an OL or CP
78-06	Defective Cutler-Hammer, Type M Relays With DC Coils	5/31/78	All Power Reactor Facilities with an OL or CP
78-07	Protection afforded by Air-Line Respirators and Supplied-Air Hoods	6/12/78	All Power Reactor Facilities with an OL, all class E and F Research Reactors with an OL, all Fuel Cycle Facilities with an OL, and all Priority 1 Material Licensees
78-08	Radiation Levels from Fuel Element Transfer Tubes	6/12/78	All Power and Research Reactor Facilities with a Fuel Element transfer tube and an OL.
78-09	BWR Drywell Leakage Paths Associated with Inadequate Drywell Closures	6/14/79	All BWR Power Reactor Facilities with an OL or CP
78-10	Bergen-Paterson Hydraulic Shock Suppressor Accumulator Spring Coils	6/27/78	All BWR Power Reactor Facilities with an OL or CP

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
78-11	Examination of Mark I Containment Torus Welds	7/21/78	BWR Power Reactor Facilities for action: Peach Bottom 2 and 3, Quad Cities 1 and 2, Hatch 1, Monticello and Vermont Yankee
78-12	Atypical Weld Material in Reactor Pressure Vessel Welds	9/29/78	All Power Reactor Facilities with an OL or CP
78-12A	Atypical Weld Material in Reactor Pressure Vessel Welds	11/24/78	All Power Reactor Facilities with an OL or CP
78-12B	Atypical Weld Material in Reactor Pressure Vessel Welds	3/19/79	All Power Reactor Facilities with an OL or CP
78-13	Failures In Source Heads of Kay-Ray, Inc., Gauges Models 7050, 7050B, 7051, 7051B, 7060, 7060B, 7061 and 7061B	10/27/78	All general and specific licensees with the subject Kay-Ray, Inc. gauges
78-14	Deterioration of Buna-N Components In ASCO Solenoids	12/19/78	All GE BWR facilities with an OL or CP
79-01	Environmental Qualification of Class IE Equipment	2/8/79	All Power Reactor Facilities with an OL or CP

LISTING OF IE BULLETINS
ISSUED IN LAST TWELVE MONTHS

Bulletin No.	Subject	Date Issued	Issued To
79-02	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	3/2/79	All Power Reactor Facilities with an OL or CP
79-03	Longitudinal Weld Defects In ASME SA-312 Type 304 Stainless Steel Pipe Spools Manufactured By Youngstown Welding and Engineering Co.	3/12/79	All Power Reactor Facilities with an OL or CP
79-04	Incorrect Weights for Swing Check Valves Manufactured by Velan Engineering Corporation	3/30/79	All Power Reactor Facilities with an OL or CP
79-05	Nuclear Incident at Three Mile Island	4/1/79	All B&W Power Reactor Facilities with an OL

NUCLEAR REGULATORY COMMISSION

OFFICE OF INSPECTION AND ENFORCEMENT

WASHINGTON, D.C.

INCIDENT MESSAGE FORM

TO: Commissioners

FROM: Lee V. Gossick, EMT

Attached is a preliminary chronology of NRC actions in connection with TMI that has been prepared in connection with the April 10 hearing.

Copies have been provided OCA, OPE and EDO offices involved.

cc: Commissioners (10) - 2 each
OPE (1)
OCA (1)
L. V. Gossick (1)
E. C. Case (2)
J. G. Davis (2)
SP - (1)
MPA (2)

DATE: 4/09/79 TIME: 1015

Contact: Steve Conver, MPA

ENCLOSURE 2

NRC REACTION TO THREE MILE ISLAND ACCIDENT

CHRONOLOGY OF EVENTS (PRELIMINARY)

Attached is a chronology of the communications and actions regarding the notification to NRC of the TMI accident, notifications made by NRC, actions taken and various related communications.

The source for the entries are as shown, and, except for the entries whose source is shown as OC Tape (telephone conversations recorded in the NRC Operations Center in Bethesda), must be treated as preliminary in nature and subject to later confirmation or clarification. It must also be noted that not all calls to and from the NRC OC were recorded due to saturation of the communications system requiring use of telephone lines not recorded by the twenty channel recorder in the Operations Center.

Date/Time	Source	Activity
Wednesday, March 28		
AM		
4:00	None	Incident sequence begins.
6:50	Operations Center (OC) tape	Licensee declares site emergency
7:02	SP Followup w/PA	Licensee notifies PEMA (Pennsylvania Emergency Management Agency) of site emergency.
7:10 - 7:45	Region I	Licensee attempts to contact Region I. Duty officer and Dep. Director en route to office when beeper sounds.
7:24	OC tape	Licensee declares general emergency.
7:45	Reg. I	Upon opening switchboard, Region I receives message from licensee
7:50	Reg. I	Region I contacts TMI control room; maintains open line.
7:55	Reg. I	Region I classifies event as a Level 1 severity incident in accordance with Region I incident response plan.
8:00	Reg. I	Region I Incident Response Center activated; John Davis, HQs., notified by Boyce Grier and also by Public Affairs (Fouchard notified by Region I Public Affairs).
8:05	Various	John Davis orders Hqs Operations Center activated. Region I State Liaison attempts to contact PA Bureau of Rad. Health.
8:20	Reg. I	Region I contacts independent measurement van (at Millstone site) and orders it dispatched to TMI site.

Date/Time	Source	Activity
8:23	OC tape	Phone call from Weiss and Moseley to Grier; John Davis calls Gossick, who was in conversation with Fouchard.
8:25	Incoming Tel. Log	Gossick calls Davis
8:30	Reg. I	PA State Police informed that NRC emergency vehicle would be en route.
8:31	OC Tape	Moseley notifies Stello & Eisenhut; Stello says he will send radiological experts to OC
8:32	OC Tape	John Davis calls L.V. Gossick, decision made to convene EMT
8:34	OC Tape	John Davis calls Denton's office; speaks to Case.
8:36	OC Tape	Davis calls Hendrie's office; speaks to Bill Dorie.
8:40 (approx)		Gossick & Case arrive at Operations Center
8:40	OC Tape	Mike Wilbur calls Boyce Grier, obtains technical information.
8:45	Reg. I	Five inspectors (including health physicists) with radiation monitoring equipment leave for site.
8:46	OC Tape	Davis calls Dorie, asks for Comm. Gilinsky in Chairman's absence; Gilinsky not in yet.
8:48	OC Tape	Dudley Thompson notifies Tom Carter, NMSS. No NMSS action required.
8:49	OC Tape	Ward notifies J. Davidson, NMSS, to make IAT notification.
8:50	Reg. I Log	Licensee calls Reg. I with current status report.

Date/Time	Source	Activity
8:50	Reg. I	State Liaison contacts PA Bureau of Rad Health
8:52	OC Tape	Davis notifies Comm. Kennedy (First Commissioner contacted) Davis reports that he, Gossick, Case & Fouchard are manning EMT at O.C. Kennedy says he will notify Commissioner Gilinsky.
8:56	OC Tape	Gossick attempts to reach Gilinsky through Bill Dorie and through Gilinsky's office. Gossick talks to John Austen and requests that Dorie reach the Chairman.
8:57	OC Tape	Davis notifies Commission Ahearne
8:59	OC Tape	Bernie Weiss calls DOE Emergency Operations Center
9:00	OC Tape	Bill Ward informs Randy Pine (CA). Randy Pine indicates that she will inform local Congressmen (Heinz & Schweiker) and Rep. __.
9:00	Reg. I	Second vehicle leaves Region for site (investigator plus inspector); Region I contacts RAP who has already been notified; two teams organized and standing by.
9:00	Bob Ryan	Ryan notified by SP Region I personnel. Arrives Hqs. OC 9:35am
9:02	OC Tape	Weiss notifies EPA (Floyd Galpin)
9:08	SP notes	Joe Fouchard calls Carl Abraham, Reg I Public Affairs

Date/Time	Source	Activity
9:10	OC Tape	Gossick calls Congressional Affairs Office. Randy Pine informs Gossick that CA had received several inquiries from local Congressmen.
9:10 to 9:30	OCA	OCA places call to majority and minority staffs of House of Subcommittee on Energy and Environment, House Subcommittee on Energy and Power and Senate Subcommittee on Nuclear Regulation as well as Senators Heinz and Schweiker and Representatives Walker and Ertel to adivse of declaration of site emergency at Three Mile Island
9:06	OC Tape	Bill Ward informs Communications Branch
9:10	OC Tape	Grier calls Moseley to explain technical aspects of incident.
9:11	Incoming Log	Gilinsky calls Gossick and Davis
9:16	OC Tape	Fouchard notifies DOE Public Affairs Office (Bob Dulin).

AFTER 9:15 AM

Other NRC personnel began arriving at Headquarters Operations Center (OC Tapes not yet scanned for times after 9:16 am)

9:15	OC Notifications	White House Situation Room Log contacted
9:27	Incoming Log	Defense Civil Preparedness Agency (DCPA) Calls Joe Hegner
10:05	PN79-67	Reg. I response team arrives at site
10:05	Reg. I	Reg I contacts EPA Hqs. (also attempts to contact EPA:III)

Date/Time	Source	Activity
10:05 to 10:10	Reg. I	Discussion w/RAP re ARMS aircraft survey. ARMS people put on standby
10:15	Reg. I	Onsite team informs PA Rad. Health that they are available for questioning.
10:16	Incoming Log	Conference call: All Commissioners and Davis
10:20	SP Log	Telephone to PA State Rad Health Dept. (Gerusky not available). Call returned at 10:45 (1st liaison established by Hqs) subsequent calls every hour or two, starting about 5:25 pm to discuss status of sampling and monitoring.
10:30 to 11:30	OCA	OCA calls principal oversight committees (including appropriations subcommittees) and Pennsylvania representatives from vicinity of site to advise of release of radioactive materials.
10:30	Reg. I	Reg. I contacts Delaware
10:40	Reg. I	Reg I contacts PA Governor's action center
10:45	Reg. I	Reg I contacts NY State Energy Office
10:30	PR#79-64	First Press Release based on Preliminary Notification
10:40	SP Log	Press Conference (from where?) patched through to PA Rad Health Dept.
10:55	SP Log	Telephone call to PA Civil Defense (second call to Operations Officer at 11:30 am).

Date/Time	Source	Activity
10:58	Incoming Log	White House Duty Officer to Weiss
11:00	Reg I	Second Team arrives at site and in control room
11:35	Reg I	FPA Region III contacted
11:45	Reg I	NJ Dept of Energy contacted
11:55	Reg I	State of MD, Power Plant Siting council contacted
PM		
12:04	Reg. I	EPA Region III contacted
12:10	Reg I	DOE (Valley Forge Office) contacted
12:30	OCA	OCA calls principal oversight committees and PA representatives regarding latest information (In response to request from Henry Myers for technical information arranged for briefing by Mr. Stello)
1:00	Reg I	Third vehicle departs for site
1:00	Reg I	Reg I contacts MD Health Dept.
1:01	Incoming Log	Stello calls Henry Myers
1:12	SP Log	SP calls VA Civil Defense
1:30	DOE Logs	DOE advance party establishes command post at Capitol City Airport
2:15	DOE Logs	ARMS helicopter arrives at site and begins tracking.
2:30	DOE Logs	BNL RAP arrives Capitol City Airport; sampling begins
2:45	Reg I	State liaison contacts Governor's office Connecticut

Date/Time	Source	Activity
3:00	SP Log	Proposed NRC Press Release cleared with PA Rad Health Dept
3:45	PN 79-67	First formal preliminary notification report distributed by NRC
4:00	OCA	Telephone briefing for Senate Subcommittee on Nuclear Regulation and Senator Heinz
5:00	PR 79-65	Second press release issued
5:15	Reg I	NRC mobile lab arrives at site
6:00	Reg I	Fourth Reg I vehicle leaves for site
7:55	SP Log	Call to PA Rad Health Dept (ARMS data shows count is up, status of BNL monitoring activities).
6:15	SP Log	PA Rad Health Dept acknowledges receipt of ARMS data and says they are keeping Governor informed.
8:17	Reg. I Log	Reg I notified of decision to send NRR team to site; arrival expected next AM.
8:30	Reg I	Fourth vehicle arrives on site
8:30- 9:45	Reg I	NRC, State, RAP Team brief Lt. Governor Scranton
9:00	Region I Log	Reg I notified that Salem providing equipment
9:00	SP Log	SP verifies (how?) that FAA has not been notified.
10:00 - 11:00	Reg. I	Reg I participates in Lt. Governor's press conference

Thursday, March 29
AM

11:30 (PM) to 12:30 (AM)	Reg I	Team briefs Governor Thornburgh
12:15	PR #79-66	Press Release
1:00	SP Log	SP notifies Defense Civil Preparedness Agency and reads press release.
2:00	SP Log	SP telephones PA Emergency Management Agency, reads press release
2:10	SP Log	SP tries to telephone Health Center for Disease Control (Atlanta) (PA radiological Health says they will try again in morning).
8:30	SP Log	SP calls H. Calley, EPA, to read press release and suggests he call Gerusky and offer assistance
9:00	Reg I	Two additional vehicles (6 people) dispatched to site
9:30	SP Log	Food and Drug Administration calls and offers to have Baltimore Field Office provide assistance in looking at food pathways
9:30		Commission briefed by Gossick, Eisenhut and Jordan at "H" Street
10:25	PN-79-67A	Issued
11:00	SP Log	SP tries to contact NY Bureau of Radiological Health (call completed at 12:17 pm)
11:05	SP Log	SP calls NJ Dept. of Health.
11:30	OCA	Chairman Hendrie and NRC staff brief members of Subcommittee on Energy and Environment, other Members of Congress, and Congressional staff on status of incident.

Date/Time	Source	Activity
11:45	SP Log	SP calls Delaware Rad-Health Dept.
12:00 (Approx)		NRR team (Vollmer +7) arrive on site; Strasma (Region III Public Affairs) on site
PM		
12:05	SP Log	SP returns call to Gov. Ray's (Washington) assistant.
12:10	SP Log	SP calls W.VA Rad Health contact
12:15	SP Log	SP calls Va. Radiological Health (call finally completed at 1:15).
12:00 - 1:00	Reg I	Vehicles 6 and 7 on site (total IE personnel: 17)
2:15	Reg I	Congressional group (Hart, Udall, Heinz, et al) arrive observation center; receive briefing.
2:40	SP Log	SP telephones Fish and Wildlife Service.
3:00	PN-79-67E	Licensee pulls thermoluminescent dosimeters from 17 fixed positions located within a 15 mile radius of site. Dosimeters had been in place for three months and had been exposed for about 32 hours after incident. Only two dosimeters showed exposures above normal levels.
3:01	SP Log	SP briefs (by phone) Md. Radiological Health on status of samples.
3:20	Reg I	State liaison contact with Vermont

Date/Time	Source	Activity
3:15 - 6:30	Reg I	NRC representatives are called to State Capital to brief Governor and Lt. Governor prior to Governor's 5:00 p.m. press Conference.
3:35	SP Log	In response to inquiry from Minnesota Rad. Health, SP gives status briefing.
5:55	Reg I Log	The Executive Management Team directs the licensee to stop dumping all water. RI notifies HQ that stopping the dumping will cause backing of water into the Turbine Building. (Late entry - the licensee has been dumping water to the river -- the water is within TS limits WRT contamination.)
6:00	Log???	PA Rad Health says NRC can make decision on rad. water dump without checking further with them.
6:10	Reg I Log	Licensee notifies NRC that he is stopping the discharge
5:30	Reg I Log	Briefings provided by Met. Edison to Senator Schweiker, Congressmen Gooding (York), Mike McCormack; Waller (Lancaster), Weidler (L.I.,NY)
6:00	PN-67B	NRC requests Met. Edison to terminate release of slightly contaminated industrial waste. Permission to resume release granted at 12:15 am and coordinated with State. State press release issued.
4:50 - 7:00	Reg. I	Multiple attempts to reach EPA:III concerning industrial waste dumps; finally contacted at 7:00 p.m.

Date/Time	Source	Activity
6:35	SP Log	PA Rad Health calls back and asks NRC to hold river dump because of Governor's concern. SP provides update on ARMS data.
7:05	SP Log	Offer from DOE Emergency Assistance.
8:30	Reg I	Call to Governor's Aide to inform of core damage
8:00 -8:30	SP Log	Updated status reports to MD, WVA and US Bureau of Rad. Health
10:05	SP Log	PA Rad Health says "go" if NRC wants to have water dumped.
10:12	Reg I Log	Industrial Waste Discharge verified to be off.
Friday March 30		
AM 12:05	Reg I Log	EMT asks IE/site to relay to Met. Ed that NRC says OK to release industrial waste. Notify NRC when release commences.
2:06	Reg I Log	Industrial Waste Tank overflowing onto ground
5:35	Reg I Log	Fire in Unit One Aux. Building Basement - (picked up from intercom). Fire in ventilation system.
8:00	SP Log	Status report to EPA
8:20	SP Log	Status report to FDA, Bureau of Radiological Health
8:30	SP Log	Status report to Md. Rad Health
8:50	SP Log	Call from FPA requesting status

Date/Time	Source	Activity
9:00	Reg I	Control room personnel (Unit 1) hear announcement that evacuation in a 10 mile radius around plant has been recommended by NRC.
9:05	Reg. I	Onsite inspector calls Region to verify that above was true. Were told recommendation was not official.
9:25	Reg I Log	State notified of release; evacuation rumored; site does not plan to call for evacuation.
9:15 - 10:10	SP Log	SP calls PA Civil Defense re potential evacuation
9:50	PN 79-67B	Issued
10:00	Reg I Log	Some confusion exists because State evidently has recommended evacuation of Middletown (Doc. Collins). Plant/NRC has not recommended evacuation.
10:25	Reg I Log	Communication Lost with Unit 2 Control room
10:30 - 10:45	Reg I Log	State has advised residents within 10 miles to go inside and shut windows. NRC position remains: no evacuation/ unnecessary to take any special precautions.
10:47	Cmsr. Tel. Log	Conference call between Gossick and Commissioners: decision to send Denton to site.
11:00	Reg. I	Unit 1 Control Room becomes aware that an evacuation was suggested by Governor.
11:40	Cmsr. Tel. Log	Hendrie and Governor discuss evacuation.

Date/Time	Source	Activity
11:45	Reg I Log	Release at 11:09 for about 15 secs.
12:03	Reg I Log	Chairman of NRC recommends that Governor of PA evacuate 5 mi. radius.
12:07	Reg I Log	EPA, Region III advised of evacuation recommendation.
12:30	Reg I Log	The NRC evacuation recommendation is changed or is clarified as follows: pregnant women and preschool children in the 5 mi. radius should be evacuated. This recommendation to Gov. PA - not public. Dr. Langford of EPA is notified of this change.
1:00	SP Log	Another conversation with FDAA re Governor's recommendation for President to call National Security Council meeting at 1:30.
1:15 - 1:30	SP Log	Calls to MD, Delaware, NY, NJ VA, W.VA Rad Health regarding Governor's recommendation.
1:25	Cmsr. Tel. Log	Hendrie conference at White House followed by 1:30 p.m. NSC meeting.
1:30	Reg I	Another vehicle (2 HPs) departs for site.
2:00	Mossburg Gossick Notes	Denton +12 arrive by helicopter at site;
2:20	Mossburg Gossick Notes	NRR Operations Center established at nearby residence; notifications to President Carter and Governor Thornburgh.
2:30	Reg I	Director and Branch Chief plus 2 HPs dispatched by helicopter.

Date/Time	Source	Activity
(About 3:30	Unverified	Wayne Kerr (OSP) +5 arrive at site to assist IE Health Physics.

By this time, 83 NRC personnel are on site and in vicinity (51 IE, 4 SP, 3 PA, 25 NRR).

Date/Time	Source	Activity
6:30	Press Release No. 79-67	Press release "no imminent danger of core melt; technical experts (Denton et al) on site"
8:45	Reg I Log	(Madden) acting as Administrative Officer reports the following: - Trailer, manned by NRR (Denton) and the White House Communications Group is now behind the Observation Center and wired for use. - Additional Trailer by 2145 will be wired with 6 telephones. - Boyce Grier is downtown in Harrisburg at a Press Conference w/Stello, Denton, Governor Thornburgh. - Air National Guard Unit at Harrisburg Int. Airport available for assistance.

Saturday, March 31

Date/Time	Source	Activity
AM 1:25	SP Log	Coordinating meeting held at Capital City Airport (EPA, DOE, PA Dept. of Environmental Resources, FDA, NRC).
3:50	SP Log	Telephone call from Bettis, Radiological Assistance Team at Command Post at Capital City Airport; analytical equipment in airport hangar.
6:00	Reg I Log	Oak Ridge man believes he can use the Loose Parts Monitoring to tell the size of the bubble in the vessel.

Date/Time	Source	Activity
6:25	SP Log	Call from PA Civil Defense; status update.
About 9:00		Public Affairs Center activated at site; limited operation until April 1.
About 8:30	SP Log	Trailer moves just outside plant gate; NRR operations center in full force.
9:17	SP Log	Call from HEW asking what their role would be if evacuation necessary; NRC says PA Civil Defense has lead; status update.
9:25	SP Log	Call from CEQ; status report
9:20	SP Log	Call from NY Rad Health Bureau; status report.
10:00	SP Log	Call from Defense Civil Preparedness Agency; status report.
PM		
12:00	SD Log	Conversations with FDA-Bureau of 12:30 Radiological Health re supplies of potassium iodine.
2:00	SP Log	PA Rad Health Dept. agrees to refer all calls relating to health matters to NRC.
3:26	Cmsr. Tel. Log	Commission meets in Bethesda at Operations Center.
4:25	Cmsr. Tel. Log	Hendrie and Governor confer via telephone on status.
5:00	SP Log	NRC informed that Governor, W. Va. had activated State Radiolcgical Assistance Team.

Date/Time	Source	Activity
Sunday April 1		
10:45	PN-67G	NRC representative (Stello?) at facility informed that sabotage attempt would be made during the night. FBI, PA State Police and licensee notified..
1:30	SP Log	SP calls DOE Command Center (they do not know where EPA people are); also calls PA Rad. Health; EPA lab is next door but no one is there.
8:13	SP Log	SP calls PA Rad Health (DOE will collect all data and transmit results to NRC). Meeting scheduled at 8:30 to set up coordination.
9:36	SP Log	Contact established by Lubenau/Vaden at PA Rad. Health offices.
11:00	Reg I Log	NRC Personnel on site (65 I&E; 27 NRR; 5 others).
	PN-67H	NRC establishes 37 TLD stations at distances from 1 to 12 miles from plant.
	PN-67H	All utilities with an operating B&W reactor are sent an NRC Bulletin to: (1) provide information on TMI-2 incident (2) require a prompt review of their plant conditions, (3) take action to prevent such an incident. NRC inspectors are being sent to each licensed B&W reactor to provide increased inspection coverage.
2:15 - 2:27	Reg I Log	President Carter is on site in Unit 2 Control Room.
8:40	Gossick Notes	NRC calls DOE/EOC to request cleanup of AUX Building... General Public Utilities requested help.

Monday April 2
AM

Date/Time	Source	Activity
3:35	Reg I Log	Hq requests licensee to send sample of containment air (2330 sample) to Bettis.
5:50	Reg I Log	Phone link drops out/HQ will try to re-establish conference call.
6:25	Reg I Log	Phone link re-established
PM Midnight	Gossick notes	Denton briefs Governor of PA.

2209

PRELIMINARY

DESCRIPTION OF EVENTS
AT THE THREE MILE ISLAND 2
FACILITY ACCIDENT

The following is a summary of the significant events that occurred at the Three Mile Island No. 2 nuclear facility on March 28, 1979, and thereafter. Attached is a detailed chronology of these events listed with the times they each occurred.

At about 4:00 am on March 28, 1979, the secondary (nonnuclear) cooling system of the Three Mile Island facility suffered a malfunction. This system normally pumps water through the plant's steam generators where the water turns to steam which then flows to turn a turbine generator. The water is then condensed back to water, is pumped by a condensate pump through a clean up system, through a feedwater pump, and finally back to the steam generators, and continually flows around this loop.

A malfunction in the main feedwater system caused the feedwater pumps to turn off (trip), which in turn caused the turbine-generator to turn off and stop generating electricity. Since the steam generators were not removing heat due to the stoppage of feedwater flow, the reactor coolant system pressure increased and the pressurizer relief valve opened to reduce reactor pressure. Immediately, the reactor turned off by the rapid insertion of the plant's control rods (scrammed) as designed and the nuclear chain reaction stopped leaving behind only residual, or decay, heat. These events all occurred within the first 30 seconds following the event.

ENCLOSURE 3

Up to this point, this sequence is normal and the auxiliary feedwater system should startup and deliver secondary coolant to the plant's two steam generators to remove heat. In addition, the pressurizer relief valve should close as reactor pressure decreases.

All three of the auxiliary feedwater pumps started but were unable to deliver flow because their flow paths were blocked by closed valves. In addition, the pressurizer relief valve failed to close and therefore allowed the reactor coolant system pressure to continue to decrease.

As the reactor pressure reached a preset value (1600 psi), the plant's Emergency Core Cooling System (ECCS) started as designed and began to inject cold water into the reactor. It is at this point that an indication of a rapidly rising pressurizer level apparently led the plant operators to terminate the ECCS flow. At this point the Three Mile Island incident had been underway for 11-12 minutes.

Between about 1 and 2 hours into the transient, the operators turned off the four large pumps which circulate the reactor coolant through the reactor. It is following this action that we believe the severe damage to the nuclear fuel began. For the next several hours there was a very large temperature difference across the nuclear core indicating little flow of coolant through the core.

During this several hour period, when severe fuel damage is occurring, primary coolant from the reactor primary coolant system was being dumped onto the reactor containment floor from flow out of the pressurizer relief valve and through the drain tank. This coolant, which contained radioactivity, was partially pumped from the reactor containment building floor to tanks in the auxiliary building. The tanks overflowed permitting radioactivity to be vented from the auxiliary building. This situation lasted until about 9:00 am when the reactor containment was sealed (isolated).

During this time, from about 6:00 am until 8:00 pm, the licensee tried to depressurize the reactor coolant system sufficiently to be able to turn on the residual heat removal system. Since his attempts failed, it was decided to repressurize the system.

After repressurization, one of the main reactor coolant pumps was restarted and flow through the reactor core was re-established.

Since feedwater was being provided to the steam generator, heat was being removed and the reactor system was slowly cooled.

Reactor cooling has essentially been in this mode since that time.

PRELIMINARY CHRONOLOGY OF THE MARCH 28, 1979 ACCIDENT AT THREE MILE ISLAND

Time (approximate)	Discussion of Events
Before 4:00 am	TMI operator working on Feedwater System.
4:00 am	The loss of all (main and auxiliary) feedwater flow occurred while the reactor was operating at 98% power. The transient was initiated by a loss of condensate pumps. The turbine tripped.
3-6 sec later	An electromatic relief valve opened to relieve pressure in the RCS* (2255 psi).
9-12 sec later	The Reactor tripped on high RCS pressure (2355 psi) to terminate the nuclear reactor and reduce power generation to decay heat alone.
12-15 sec later	The RCS pressure decayed to the point (2205 psi) where the relief valve should have reclosed. The RCS continued to depressurize for about the next two hours.
15 sec later	The temperature in the RCS hot leg peaks at about 610^{o}F with a pressure of about 2150 psi.
30 sec later	The auxiliary feedwater pumps in both safety trains (1 turbine driven pump and 2 electrically driven pumps) were started and were running at pressure ready to inject water into the steam generators and remove the residual heat produced in the reactor core. No water was injected since the discharge valves were closed.

*Throughout, RCS denotes "reactor coolant system."

Time (approximate)	Discussion of Events
4:01 am	The pressurizer level indication began to rise rapidly. The steam generators, A and B, had low levels of water and were drying out.
4:02 am	The ECCS was initiated as the RCS pressure decreased to 1600 psi.
4:04-4:11 am	The pressurizer level indication went offscale high and the operator manually tripped the first HPI pumps at about 4:04:30 and the second at about 4:10:30.
4:06 am	Water in the RCS flashed to steam as the pressure bottoms out at 1350 psi. The hog leg temperature was about 585^oF.
4:07-4:08 am	The Reactor building sump pump came on.
4:08 am	The operator opened the valves at the discharge of the auxiliary feedwater pump allowing water to be injected into the steam generators.
4:11-4:12 am	The operator restarted the ECCS to inject water into the RCS to control pressurizer level.
4:11 am	The pressurizer level indication comes back on scale.
4:15 am	The RC Drain (Quench) tank rupture disk blew at 190 psig due to continued discharge of the relief valve that had failed to open.
4:20-5:00 am	The RCS parameters stabilized at a saturated condition of about 1015 psi and 550^oF.
5:15 am	The operator tripped both RC pumps in Loop B.
5:40 am	The operator tripped both RC pumps in Loop A.

Time (approximate)	Discussion of Events
5:45-6 am	The reactor core began a heatup transient. The RCS hot leg temperature went offscale at 620 degrees F within 14 minutes and the cold leg temperature dropped to near the temperature of high pressure injection water (150 degrees F).
6:20 am	The failed open relief valve was isolated by the operator by closing a block valve. The operator also isolated steam generator B to prevent leakage of radioactive secondary water from leaking S.B. tubes.
7:00 am	The RCS pressure had increased to 2150 psi and the relief valve was opened to relieve RCS pressure.
7:15 am	A pressure spike of 5 psig occurred in the RC drain tank due to steam from the relief valve.
7:45 am	A pressure spike of 11 psig occurred in the RC drain tank and the pressure in the RCS was at 1750 psi.
9:00 am	The pressure in containment peaked at 4.5 psig.
9:00-11:00 am	The RCS pressure increased from 1250 psi to 2100 psi.
11:30 am	The operator opened the pressurizer relief valve to depressurize the RCS in an attempt to initiate RHR cooling at 400 psi.
12:00 am - 1:00 pm	The RCS pressure decreased to about 500 psi and the core flooding tanks partially discharged. The relief capacity was not sufficient to vent enough to reach 400 psi.
2:00 pm	The pressure in the containment spikes at 28 psig causing containment sprays to be initiated. The operator stopped the spray pumps after about 2 minutes of operation.

Time (approximate)	Discussion of Events
5:30 pm	The pressurizer relief valve was closed in order to repressurize the reactor coolant system.
5:30 - 8 pm	The RCS pressure increased from 650 psi to 2300 psi.
8 pm	RC pump in Loop A was started at which time the hot leg temperature decreased to about 560 degrees F and the cold leg temperature increased to 400 degrees F, indicating flow through the steam generator. Thereafter, the reactor was being cooled by reestablishing condenser vacuum and steaming to the condenser by steam generator A with the RCS cooled to about 280 degrees F and 1000 psi.

March 29

The RCS temperature and pressure was stablized at about 280 degrees F and 840 to 1020 psi. The maximum reading on the incore thermocouples was 612°F, but several were not with range for computer readouts (printing "?") which was subsequently found to indicate greater than 700 degrees F.

March 30

The RCS temperature and pressure was stable at nearly 280 degrees F and between about 1000 to 1060 psi. Several incore thermocouples were beyond the range for computer readout, the maximum indicated reading was 659 degrees F. The NRR staff estimated the bubble size in RCS to be about 1200 ft^3 and requested the licensee to refine their calculation of the bubble size.

March 31

The RCS temperature and pressure remained stable at about 280°F and 1000 psi. Slight drop in pressurizer level 251-191". Temperatures in the core as measured from the incore thermocouples were gradually decreasing (maximum indicated about 500°F). The hydrogen recombiner was in an operable status but additional shielding was needed and was being obtained. Two samples of containment atmosphere were analyzed which showed a hydrogen concentration of 1.7% and 1.0%. Licensee calculated bubble size to be about 620 ft^3 @ 875 psig.

April 1

No substantial change in RCS temperature and pressure
Incore thermocouples continue to show decreased trend.

Licensee continued hookup of hydrogen recombiners and addition of shielding. Licensee calculated valves of bubble size varied. Containment air samples indicate 2.3% hydrogen.

April 2

Reactor pressure stable at about 1000 psi. Incore thermocouples continued to show a decrease with all measurements below 475°F. Inlet and outlet temperatures were still about 280°F. One hydrogen recombiner was put in operation.

Analysis indicated that the oxygen generation rate in reactor less than originally estimated. Measurements indicated that the bubble was being significantly reduced.

April 3

Reactor pressure and temperature stable at 1000 psi and 280°F, respectively. Thermocouple readings analyzed- maximum 477°F, only 3 thermocouples were above 400°F. Gas bubble size much reduced. Containment about 1.9% hydrogen. One pressurizer level indicator failed.

April 4

Reactor pressure and temperature stable at 1000 psi and 280°F, respectively. Thermocouple maximum temperature was 466°F. Gas bubble size decreasing. Vent valve on pressurizer intermittently opened and degassing continues through letdown system.

April 5

Reactor pressure and temperature stable at 1000 psi and 280°F, respectively. Maximum thermocouple reading is 462°F. Pressurizer level responding normally to pressure changes indicating a completely full system.

Containment atmosphere indicates 2% hydrogen. One recombiner operating, one in standby. Pressurizer vented to containment about 15 minutes every 6-8 hours.

April 6

Reactor pressure stable at about 1000 psi and temperature about 285°F.

At approximately 1:25 pm, reactor coolant pump 1A tripped and reactor coolant pump 2A was started within about 2 minutes. Shift in thermocouple readings. The three thermocouples previously reading about 400°F are presently reading between 285°F and 315°F. Central thermocouple increased from 375°F to 425°F and is the only one reading about 400°F.

Containment measurements indicate about 2% hydrogen. Pump-back system for pumping waste gas decay tank volume to containment began.

April 7

Reactor pressure and temperature stable at about 1000 psi and 280°F, respectively.

At about 8 pm, the licensee began to slowly lower reactor system pressure. The slow decrease will end when reactor pressure reaches 500 psi. This is a step toward cold shutdown and includes degasification to prevent bubble formation as pressure and temperature decreases.

Hydrogen concentration in the containment is about 1.9%.

GENERIC CONSIDERATIONS

OF

TMI-2 INCIDENT

The following are the significant sequence of events that occurred at TMI-2.

a. The turbine tripped due to loss of main feedwater,

b. The reactor tripped,

c. The auxiliary feedwater pumps started but flow was not auxomatically established,

d. The pressurizer relief valve apparently stuck open,

e. The high pressure injection was turned off, and

f. The reactor coolant pumps were turned off.

Our preliminary evaluation indicates that the incident may have been compounded by misleading indication of primary system water level. In addition, the consequences of the incident were increased by the lack of prompt automatic containment isolation.

The initiating event, i.e., turbine trip and subsequent reactor trip, are anticipated events in that they are expected to occur during the plant lifetime and the system is designed to respond safely. In fact, other B&W designed operating plants have experienced these kinds of transients and have responded safely. As a result of our preliminary evaluation of the TMI incident, however, we have preliminarily identified several human, design, and mechanical failures. They are all essentially related to the loss of feedwater (item c above), the turning off of the high pressure injection, (item e above), and the turning off the reactor coolant pumps (item f above). To ensure that these potential human, design, and mechanical failures do not result in a similar incident at other operating facilities, we have directed (via IE Bulletins) owners of facilities with B&W reactors to take several steps to ensure that safety margins are maintained. In addition, we have formed an NRC Task Force to review in detail the causes of the TMI-2 incident and upon completion of these efforts will take subsequent actions as appropriate. The Task Force report will be completed about the end of this month. These NRC actions are also being taken at this time because of the preliminary nature of our evaluation. Certain additional information will be developed which will provide additional insights into the actual causes and consequences of the various actions during the event. At this time, however, our preliminary understanding of the event is sufficient to enable us to define the immediate actions required of operating facilities with B&W reactors to prevent such an occurrence at these plants, and provide us with an adequate basis to allow continued operation of these facilities.

First, the incident at TMI-2 was initiated by a loss of auxiliary feedwater following a turbine trip (item c above). Since plants are not designed and evaluated for the complete loss of all feedwater, we have taken steps to ensure that the emergency feedwater system will be available to inject water under this situation. At TMI-2, the block valves in the discharge lines from the auxiliary feedwater pumps were closed. We have required that operating facilities with B&W

reactors ensure that these valves are always open by requesting their specific examination of these valve positions. The position indication will be further verified by a full-time NRC IE inspector at each of these plants.

Another generic aspect of this event and a significant contributor was the apparent sticking open of the electromatic valve on the pressurizer. Licensees are being requested to examine their procedures such that operators are aware all valve positions, including the backup block valve to the relief valve, and have information available to permit its use. The sticking open of the relief valve was a significant contributor to this event and would be considered an important event regardless of whether auxiliary feedwater is available or not.

The second significant concern, which also has generic considerations, concerns the turning off of the high pressure injection system. In general, we have requested all operators of plants to exercise extreme caution before turning off any safety system. Specifically, we have taken steps to require operators to maintain high pressure injection for a minimum of 20 minutes if it is automatically actuated. This occurs on low reactor pressure (1600 psi) in the reactor. We further require that high pressure injection be maintained until stable conditions are obtained. We believe such actions may cause operational inconveniences, but that they are not significant when compared to the gain to be made should a severe transient occur. We also require that the LPI be maintained for 20 minutes following any low pressure transients, including the case where a relief valve inadvertently opens and sticks open, to ensure pump coolant inventory.

Finally, we are requiring that if the reactor coolant pumps are in operation when a severe feedwater transient might occur, they should be kept in operation if at all possible. Furthermore, if possible, one should be kept running in each loop. This requirement provides an extra level of safety to cover a broad variety of transients. In developing this requirement, it was recognized that operation of the RCPs under certain conditions may damage the pump due to cavitation, however, it is believed that such operation is appropriate to ensure adequate response to a wide variety of transients.

The above mentioned staff requirements, in addition to our requirement that all licensees with B&W reactors review their designs, have been imposed to maintain and possibly increase margins regarding their response to feedwater, and other, transients. Such actions will, we believe, compensate for any remaining generic concerns regarding B&W reactors response to such transients.

Because the accident situation appears to have been further complicated by the containment not being isolated upon ECCS actuation, (in this case HPI), we have also taken steps to ensure that the containment is isolated to the extent possible given any particular event.

The above mentioned considerations have been directed towards B&W reactors because they appear to be the ones most directly affected. The described actions are intended primarily to be short term actions and may well be modified as a result of the NRC Task Force review of B&W reactor transients. Certain of these interim actions may also later be shown to be applicable to other pressurized water reactors, i.e., those designed by Westinghouse and Combustion Engineering. These facilities have significant design differences. Two significant differences between the B&W primary system design and those of Westinghouse and Combustion Engineering are, first, the primary system water level for Westinghouse and Combustion Engineering are more directly measured. The operators would, therefore, have had a more direct indication of primary system water level and would have been so influenced before stopping high pressure injection flow. Secondly, the steam generator volumes are larger for Westinghouse and Combustion Engineering plants, and therefore are less sensitive to feedwater transients allowing more time to detect and correct any deficiencies in auxiliary feedwater flow. Therefore, we have not required any actions of licensees with Westinghouse and Combustion Engineering designed plants at this time, although we have been sending them information copies of all actions required of owners of B&W reactors.

Three Mile Island Incident

Summary of Initial Response and Radiological Surveys

At about 0700 hours, the license identified high levels of radioactivity in the reactor coolant sample lines - there were radiation readings of about 600 mr/hr at contact with the sample lines - and a "site emergency" was declared.

At 0730 hours the license declared a "general emergency" based on high radiation levels in the reactor building, and began notification of certain Federal, State and local agencies according to emergency procedures. At 0830 hours the radiation levels at the site boundary were reported to be less than 1 mr/hr.

NRC Region I received notification of the general emergency condition at the plant at approximately 0745 hours, March 28. After evaluation of the reported conditions, an NRC incident response team was assembled and dispatched to the site and the situation was reported to NRC Headquarters.

The NRC response team, consisting of reactor operations specialists and health physicists, left the Region I Office at 0845 hours and arrived onsite at 1005 hours, March 28.

Concurrent with the assembly and dispatch of the team, operations centers were activated both at the Region I office and at NRC Headquarters. Notificaton procedures were initiated at both NRC Region I and NRC Headquarters to inform the Commissioners, NRC staff and other State and Federal agencies.

At 0859 hours the NRC notified the Department of Energy's Emergency Operations Center at Germantown, Maryland and requested that an aerial survey (AMS, Aerial Measurement System) team be dispatched promptly to the site. The AMS helicopter arrived at the site and had located, tracked and made measurements in the plume by 1515 hours, March 28.

Returning to earlier events, the NRC incident team, arriving onsite at 1005 hours, measured radiation levels of less than 1 mr/hr at the north gate, 3 mr/hr in the north parking lot and 7 mr/hr at the east side of the island. The NRC team, after being briefed by the licensee regarding radiological and plant conditions, immediately set out to gather additional radiological data.

Radiation monitors in the plant showed abnormally high radiation levels in the containment and auxiliary building which prevented personnel access into certain areas. Radiation surveys also identified elevated levels of radiation outside plant buildings; however, the condition outside were not of such a level to prevent the gathering of survey data. Radiation surveys determined that a release of airborne radioactivity was occurring.

At 1110 hours, radiation levels of 3 mr/hr were measured at the plants' observation center on Route 441 immediately east of the plant, and at 1130 hours, levels of 0.3 mr/hr were measured on Route 283 near Harrisburg.

By the afternoon of March 28, survey measurements showed radiation levels up to 15 mr/hr (beta-gamma) in the plume at ground level and levels generally less than 1 mr/hr (beta-gamma) outside the plume. The highest measurement of about 70 mr/hr (beta-gamma) was at 1620 hours at the north gate of the plant.

Aerial surveys the afternoon of March 28 located the plume travelling in a N to NE direction in approximately a 30° sector, and radiation levels of 0.1 mr/hr were measured at about 16 miles from the site at an altitude of several hundred feet. The aircraft survey identified, by gamma spectral analysis, the radioactivity as principally xenon-133.

By the evening of March 28, the agencies conducting radiation surveys and sampling operations included the licensee, NRC, DOE and the State of Pennsylvania. In addition to the DOE AMS helicopter and aerial survey team and the NRC team and portable equipment, a mobile laboratory of the NRC Region I office had arrived at the site to process and analyze samples.

Sampling and analysis of milk and air sampling for radioiodine had begun and has continued to date. Thus far, only low levels of radioiodine in milk and air have been reported. The levels are far below the level of action for control of dairy herds or milk. The sampling will continue until some time in the future and the results are continuing to be evaluated.

By March 29, the NRC team at the site had established a routine operation and procedures for obtaining both onsite and offsite radiological data. This information was being relayed to the NRC Region I (Philadelphia) office and to the NRC operations center in Bethesda, Maryland. Aerial surveys were being conducted at 3-6 hour intervals.

During March 29, radiation levels at the site boundary on the island ranged up to about 50 mr/hr (beta-gamma). The plume during the morning of March 29 extended in a N to NW direction, and aerial surveys measured 0.5 mr/hr at 1 mile and 0.2 mr/hr at 10 miles from the site. Offsite ground surveys measured levels generally less than 1 mr/hr during the day; maximum offsite radiation levels of 20 mr/hr (gamma) and 30 mr/hr (beta/gamma) were measured one mile west of the plant in Goldsboro at 0600 hours. These levels persisted for a short period of time, less than one hour.

By the end of March 30, the number of NRC staff at the site had grown to 83, including Regional Office and Headquarters personnel. Radiation surveys were more scheduled and routine. Ground level surveys in offsite areas downwind from the site measured radiation level ranging from less than 0.1 up to 1.8 mr/hr. Aerial surveys measured radiation levels of 8-10 mr/hr over the site, and levels of 6-8 mr/hr in the plume near the site. In the evening of March 30, the plume was tracked in a northwesternly westernly direction from the site and was not detectable beyond 5-6 miles away.

By the end of March 31, ground and aerial surveys were being coordinated on a frequent scheduled basis and the results being reported regularly to NRC Headquarters. Information on results of milk, water and air sampling was being received and evaluated. Results of licensee's TLD stations (18 stations within a 15-mile radius of the reactor) were received. The TLDs had been in place for three months and had been exposed for about 32 hours after the incident. Three dosimeters showed exposures above normal levels; the highest was from a station on Three Mile Island, 0.2 miles NNW of the reactor - 921 mr, approximately 905 mr above previous normal quarterly readings; the other high readings were 0.4 miles north of the reactor -81 mr, approximately 65 mr above previous normal quarterly readings and a station at north bridge, 0.7 miles NNE of the reactor - 37 mr, about 22 mr above normal quarterly readings.

On March 31, the NRC established 37 TLD stations within a radius of 12 miles of the site. Two or more dosimeters were placed at each station, one to be left indefinitely for integrated dose and the others to be changed daily. The first day of this monitoring (March 31-April 1) showed the highest reading of 1.1 mr/hr at ½-mile ENE of the plant. Other readings were much less and have decreased steadily since.

As of April 4, the following information had been received on sampling and analysis for radioiodine:

Approximately 130 offsite water samples, analyzed by NRC, DOE and the Commonwealth of Pennsylvania, showed no detectable radioiodine.

Approximately 150 offsite air samples had been taken and analyzed by NRC, DOE, the licensee and the Commonwealth. Samples were collected at distances out to 40 miles. Only 8 of the samples indicated detectable concentrations of iodine; these were in the range 2.7×10^{-13} to 2.4×10^{-11} microcuries/cc, the highest being about 1/4 of the MPC established for unrestricted areas in 10 CFR Part 20.

Approximately 200 samples of milk had been analyzed by the State and FDA. The results ranged from minimum detectable activity to 41 picocuries per liter; there were two samples at or near the higher level. By comparison, the HEW recommends placing dairy herds on stored food when iodine-131 in milk reaches 12,000 picocuries per liter.

Approximately 170 vegetation samples had been collected and analyzed by DOE, NRC and the Commonwealth of Pennsylvania. The samples were collected from various sites within 2 miles of the plant. None showed any detectable radioiodine.

Approximately 150 samples of soil were collected and analyzed by NRC and DOE. None showed any detectable radioiodine.

As stated previously, sampling and analysis of air and milk for radioiodine is continuing. As a further measure in evaluating the significance and health implications of any radioiodine released from the plant, actions have been initiated to have a selected number of persons analyzed in a "whole-body", radiation measuring system. The people selected would include both licensee employees who were onsite, and local offsite residents.

Recent aerial and ground level surveys results indicate radiation exposure rates to be consistently less than 0.1 mr/hr.

SUMMARY OF RADIOACTIVE LIQUID

RELEASE SITUATION

(IWTS)

The Industrial Waste Treatment Sump (IWTS) and the Industrial Waste Filter Sump (IWFS) normally collect nonradioactive liquid industrial wastes at the TMI facility. The normal sources of water to these sumps are floor drains and other sumps located in facilities which do not have radioactive systems. The IWFS and IWTS are periodically discharged to the Susquehanna River by being pumped (approximately 130 gpm) into the cooling tower blowdown which flows into the river at a location just South of the Unit 2 mechanical draft cooling tower. The 60,000 gallon per minute cooling tower blowdown dilutes the IWFS/IWTS discharge by a factor of approximately 500 before it enters the river.

During the TMI incident, the high concentration of radioactivity in primary systems cross-contaminated normally non-radioactive secondary systems and contaminated water eventually ended up in the IWFS/IWTS. The precise timing of these events is not known. To preclude overflow of the IWFS/IWTS the licensee initiated discharge at about 1300 hours on Thursday, March 29. Log records show that the licensee considered a controlled diluted release a better alternative than letting the sump overflow with a resulting undiluted release to the river. NRC analysis of water in the IWFS/IWTS at this time indicated the presence of Xe-133 and Xe-135, but results were not precisely known because the radioactivity in the gaseous plume being released interfered with radiation counting instruments used to analyze the samples. However, it was determined that no iodine was present and the Xe concentrations were not considered significant for the release pathway. Prior to the initial release NRC consulted with the State of Pennsylvania, Bureau of Rad Health who was in agreement with NRC's position to allow releases as long as Technical Specifications were met. The State later, however, expressed concerns and at approximately 1800 hours on Thursday, March 29, NRC requested the licensee to stop discharging pending further analysis of the situation.

After consultation with the State, NRC and Pennsylvania subsequently authorized the continuation of release at 0015 hours on Friday, March 30. The licensee began making releases again at 0430 hours on Friday, March 30. Intermittant releases continued until Monday, April 2, when the State of Pennsylvania requested they be discontinued.

Due to a backlog of samples for analysis, NRC did not become aware of radioiodine in the waste water until the sample counted at 1430 hours on Saturday, March 31; this sample had been collected at 0400 hours that morning. Also, because samples were being split with the licensee's contractor, the samples were not being analyzed in the same order in which they were collected. During subsequent analysis it was determined that iodine was initially present in the discharge on early Friday, March 30, as identified by a sample collected at 0200 hours on Friday, March 30. This particular sample was analyzed at 0133 hours on Sunday, April 1. Releases were again terminated at 1110 hours on Monday, April 2. Several reasons exist for the approximate two day time frame between identification of radioiodine in the water and cessation of releases. First, the NRC laboratory was merely processing samples and was not in a position to interpret the results. Second, the feedback of results of sample analyses to those who might interpret them was hampered by the overall urgency of the moment. Third, the majority of samples were below the release limit, thereby making this a low priority matter relative to the other events at the time.

After further review of the matter and additional consultation with both the States of Pennsylvania and Maryland, joint agreement was reached and the licensee was again authorized to make releases on Thursday morning 4/5. The licensee began discharging again at about 0300 hours on Friday with no further problems noted.

It should be noted that three organizations have been performing sample analysis. NRC and the licensee's contractor were performing analysis on the scene and the samples were then forwarded to the State of Maryland for analysis. Analyses performed at the scene were considered preliminary because of counting interference caused by the gaseous releases from TMI

Unit 2. Obviously, there would be some time delay in receiving Maryland's results.

It should also be noted that a State of Pennsylvania representative (Bill Dornsife) was at the scene throughout these events and was frequently receiving the results of all sample results. This person stated that he had received all information in a timely manner.

Review of sample analysis indicates that the maximum concentration of radioiodine released was 2.7×10^{-6} microcuries per milliliter (μCi/ml) of iodine-131, after dilution; this is approximately nine times the instantaneous release rate limit specified in the plant's Technical Specifications; the limits are the concentrations for unrestricted areas listed in Table 2, Appendix B, 10 CFR Part 20 and whereas Part 20 allows concentrations to be averaged over a year to achieve compliance, the Technical Specifications limit discharges to Part 20 concentrations on an instantaneous basis. The majority of samples averaged approximately 2.4×10^{-7} μCi/mℓ or about 80% of the Technical Specification limit. At no time was any radioiodine identified in any river water samples collected downstream of the release point.

PRELIMINARY EVALUATION OF HEALTH EFFECTS
OF THE THREE MILE ISLAND INCIDENT

Radiation monitoring indicates that the exposure of the general population in the immediate vicinity of the plant was well within the limits of NRC regulations (10 CFR Part 20) for annual doses to members of the general populations. They did exceed the numerical design objectives for normal reactor operation of 5 millirem per year (Appendix I to 10 CFR Part 50). The sources of exposure were radioactive gases (xenon, krypton and iodine) that leaked from the plant -- primarily from the auxiliary building. Radioactive iodine (I-131) would be of particular concern because of its concentration in food, particularly in milk. However, as of April 3, 1979, it appears that no more than 3.0 curies of radioiodine were released. Iodine levels observed in milk samples are less than one-tenth of those observed in milk following the Chinese nuclear tests in the fall of 1977. The predominant radioactivity released from Three Mile Island was the noble gas Xenon-133. The NRC has estimated that as of April 5, 1979, approximately 10 million curies of Xe-133 were released. An independent estimate of Xe-133 releases by Lawrence Livermore Laboratory is 14 to 34 million curies, thus both estimates are in agreement considering the extent of the uncertainty in the estimates.

An ad-hoc dose assessment group of representatives from NRC, EPA and HEW have made estimates of the radiation doses to the population around the Three Mile Island plant, based primarily on monitoring of offsite areas by thermoluminescent dosimeters. The calculated, total cumulative, 50-mile radius population dose from March 28 to April 8,* was approximately 2400 man-rems, which is equivalent to an average dose to individuals of 1.1 millirems. The maximum dose to an individual offsite (hypothetical individual continuously present out-of-doors at a location 0.7 miles NE of the plant) is still estimated to be less than 100 millirems (85 millirem). This is within the dose limits recommended by the National Council on Radiation Protection and Measurements for annual doses to a member of the general public (170 millirem per year). These estimates are whole body gamma doses resulting primarily from the relatively weak gamma radiation from Xenon-133 (80 keV).

The health impact of the estimate of maximum individual dose for a hypothetical individual exposed out-of-doors for entire duration at location of highest measured offsite dose can be considered in terms of the added risk of a fatal cancer. The existing lifetime risk of fatal cancer is approximately one-eighth (0.12). The estimated risk from natural background is approximately one to two percent of this value (0.0017). The added risk delivered to the hypothetical individual would be 0.16% (0.000019) of the existing risk of a cancer death or about one percent of the estimated risk from natural radiation (1.1%). The potential health impact of the estimated population dose including fatal and non-fatal cancers and genetic effects to all future generations is 1.3 health effects; and the number of potential fatal cancers over the lifetime of the population is 0.45. This can be compared to the existing cancer death rate of 4,500 per year and

*From April 4th on, these values have been updated by members of the NRC Staff.

the estimated incidence from natural background radiation (125 millirem per year) of 54 per year. This supports the conclusion that the accident will not produce any detectable cancers within the lifetime of all of the residents in the area.

Resident Inspection Program

In May 1977 the NRC decided to adopt a revised inspection program that, when fully implemented, will include assignment of resident inspectors to each operating power reactor site, to selected fuel facilities and to power reactor sites in the later stages of construction. This decision was based upon the results of a two-year trial program of resident inspection which was completed in October 1976. The trial program demonstrated that the concept of locating inspectors near reactor sites has the potential for increases in both effectiveness and efficiency when compared to the program of inspections conducted by inspectors based in a regional office which may be several hundred miles from a reactor site.

In May 1977, the Commission requested that OMB approve an amendment to IE's FY 1978 budget to provide resources needed to begin implementation of the revised inspection program. A FY 1978 supplemental request was submitted to the OMB on September 15, 1977 and was signed by the President on September 8, 1978. The initial resident inspector under this program arrived onsite in July 1978. Currently, there are resident inspectors assigned to 20 reactor sites. As a result of the Three Mile Island accident, at each of these sites with similar Babcock and Wilcox designed reactors, a full-time inspector was assigned to provide the equivalent of the resident inspection program.

A description of the Revised Inspection Program for Nuclear Power Plants was published as NUREG-0397 (March 1978).

The four elements of the revised inspection program provide a balanced examination of the activities of the licensee. The revised program consists of:

- Resident inspectors onsite at all reactors in operation, at reactors in the late stages of construction, and at selected fuel facilities.

- Region-based inspectors who will supplement the inspections performed by the residents with highly specialized inspections in such areas as environmental monitoring, physical security and health physics.

- Performance appraisal inspectors to independently assess licensee performance, the effectiveness of the NRC inspection program and to confirm the objectivity of inspectors.

- Increased independent verification of licensee activities.

The specific requirements of resident, specialist, and performance-appraisal inspectors and of the entire program create a need for a comprehensive approach to training, management of the inspection force and career development. An expanded training program for all types of inspectors and an enhanced career management plan assure the performance of all elements of the inspection program by well trained and experienced inspectors.

In May 1977, the Commission requested that OMB approve a budget amendment for IE for 125 people and $6,000,000. These resources were requested to allow IE to begin implementation of the resident inspection program in FY 1978 with full implementation in FY 1981.

In June 1977, the Office of Inspection and Enforcement developed its program plan for implementation of the resident inspection program. This plan assumed timely approval of the budget amendment request and aimed toward having the first resident inspectors onsite in FY 1978 with full implementation by the end of FY 1981. The planned schedule was:

	FY 78	FY 79	FY 80	FY 81	FY 82
Residents	35	46	111	133	152

In September 1977, the OMB approved a portion of the May 1977 request for additional resources. This OMB approval provided for increasing the NRC full-time personnel ceiling by 75 to allow earlier implementation of the revised program. Under this OMB-approved amendment (submitted to Congress in January 1978), the manning schedule was:

	FY 78	FY 79	FY 80	FY 81	FY 82
Residents	22	49	76*	93	98

*All operating reactor sites manned.

At full implementation, it was planned to have at least one inspector at sites with reactors in the later stages of construction, in pre-operational test, or in operation.

In September 1978, the FY 1978 supplemental request was signed by the President and the NRC was allocated 61 positions and $2,650,000 for the revised inspection program. The Office of Inspection and Enforcement, in anticipation of approval of the FY 1978 supplemental had initiated recruitment efforts and was successful in promptly recruiting personnel.

In the planning for the resident inspection program, an integral part of the revised program was a national level performance appraisal effort. This is intended to provide for (1) evaluation of NRC licensee performance from a national perspective, (2) evaluation of the effectiveness of the NRC inspection program, and (3) confirmation of the objectivity of NRC inspectors.

The modifications to the revised inspection program from the initial budget amendment request to OMB to the approved FY 1978 supplemental are:

Budget amendment request for 125 additional people (May 1977)

Manning Schedule:

	FY 78	FY 79	FY 80	(Full Implementation) FY 81	FY 82
Resident Inspectors	35	46	111	133	152
Performance Appraisal Inspectors	42	78	119	157	157

Full implementation in FY 1981 includes:

- One inspector for each reactor phase (construction, preoperational testing, operations) at a site.
- Additional inspectors so that no inspector would cover more than two reactors in any one phase.

Approved FY 1978 supplemental of 61 additional people

Manning Schedule:

	FY 78	FY 79	FY 80	(Full Implementation) FY 81	FY 82
Resident Inspectors	22	49	76	93	98
Performance Appraisal Inspectors	10	15	17	28	33

Full implementation in FY 1981 includes:

- One resident at each site with an operating reactor
- One resident at each site with a reactor in a later stage of construction
- One resident at each of six selected fuel plants

Currently, there are resident inspectors assigned to 20 reactor sites. At these 20 sites are:

26 operating reactors

2 reactors in the preoperational test phase

6 reactors under construction

As a result of the Three Mile Island accident, at each of three additional sites, an inspector was assigned to provide the equivalent of the resident inspection program. On these three sites is a reactor of Babcock and Wilcox design similar to the Three Mile Island reactor.

The phased implementation of the resident inspection program resulted from the need to train newly hired inspectors, and the need to maintain a qualified base of inspectors in the regional offices. Currently, the implementation of the program is on schedule.

POPULATION DOSE ESTIMATES

During the week of April 1, a joint NRC/HEW/EPA ad-hoc study group agreed on the methodology to be used in estimating the radiation dose received by the population within a fifty (50) mile radius of the site. In addition, the study group calculated the initial dose estimates up to April 4. Using the agreed upon methodology, as of noon April 8, the NRC has estimated the total population dose within a fifty mile radius to be 2400 man-rem. There are approximately 2 million people living within the fifty mile radius of the site. Thus, the radiation dose to an average member of the population is estimated to be in the range of 1-2 millirem.

It is estimated that the maximum radiation dose received offsite by a member of the public is less than 100 millirem (~85 millirem is the current best estimate). This individual would have had to be continuously present out-of-doors at the site boundary approximately 0.7 miles northeast of the reactor, which is the point at which the higher radiation dose rates were measured.