DEPARTAMENTO DE TECNOLOGIA DE REATORES | DATA 1/12/80

DIVISÃO DE | 0 911

DETR.PD-072/79

TITULO OCORRÊNCIAS RELACIONADAS COM A SEGURANÇA EM CENTRAIS NUCLEARES

ADENDO 1 À NOTA TÉCNICA DETR.PD 119/80, DE 3.9.80.

NOTAS CORRELATAS

DETR.PD 119/80

OBJETIVO

Complementar e atualizar a Nota Técnica "Ocorrências Relacionadas com a Segurança em Centrais Nucleares" publicada em 3.9.80.

LISTA DE DISTRIBUIÇÃO

SUPED	(1)
ASPC.PD *	(1)
DETR.PD	(2)
AUTORES	(3)
SEDOTE.PD	(1)
Eq.Anal.Acid.	(1)

* Apenas folha de rosto

RESUMO E CONCLUSÕES

São apresentadas algumas ocorrências relacionadas com a Segurança em Centrais Nucleares a água leve dos EUA, selecionadas devido a suas singularidades e/ou ao seu interesse nas operações de reatores, publicadas na "NUCLEAR SAFETY" no período de maio a agosto/80.

Devido a sua importância, foram incluídos alguns artigos publicados na "NUCLEAR SAFETY", sobre o acidente de "Three Mile Island-2".

ÍNDICE

mmcd/.

Nº. CÓPIAS 9

AUTOR (ES)	VISTO	DATA	APROVAÇÃO	VISTO	DATA
Carlos V.G.Azevedo	[assinatura]	1/12/80	CHEFE DO LAB. OU GRUPO		
Luiz F.Sirimarco	[assinatura]	1/12/80	p/ CHEFE DA DIVISÃO	[assinatura]	03/12/80
Walkírio R.A. Lavorato	[assinatura]	1/12/80			

CLASSIFICAÇÃO

TAREFA: 11.28

OCORRÊNCIAS RELACIONADAS COM A SEGURANÇA EM CENTRAIS NUCLEARES

1. INTRODUÇÃO

Nesta Nota, visa-se complementar e atualizar a Nota Técnica "Ocorrências Relacionadas com a Segurança em Centrais Nucleares" publicada em 3.9.80, onde são apresentados alguns eventos ocorridos em Centrais Nucleares a Água Leve, nos Estados Unidos da América do Norte.

Essas ocorrências foram selecionadas pela equipe da revista "Nuclear Safety" e publicadas no período de maio a agosto de 1980, devido a suas singularidades e/ou interesse geral na operação de reatores.

Alguns artigos sobre o acidente de "Three Mile Island-2", publicados no periódico acima mencionado, foram incluídos, devido a sua importância.

2. OCORRÊNCIAS EM PWR

Ver página 3/53.

DETR.PD

NUCLEBRÁS

DETR.PD 177/80

SELECTED SAFETY - RELATED EVENTS IN NUCLEAR POWER. PLANTS - PWR

DATE	EVENT	FACILITY	REACTOR SUPPLIER	MW (E)	ANEXO
03.28.79	THE THREE MILE ISLAND ACCIDENT	T.M.Island 2	B&W	906	
	- Preliminary Report on the TMI Incident				1
	- Developments Pertaining to the TMI Accident				2
	- Summary of TMI Lessons Learned Task Force Report				3
	- Report of the President's Commission on the Accident at TMI				4
	- The Rogovin Report on TMI				5
10.02.79	STEAM-GENERATOR TUBE RUPTURE	Prairie Island 1	W	520	6.1
02.26.80	INSTRUMENTATION AND CONTROL FAILURE	Crystal River 3	B&W	825	6.2

ABBREVIATIONS: B&W - Babcock & Wilcox Co.
W - Westinghouse Electric Co.

Preliminary Report on the Three Mile Island Incident

By W. R. Casto* and Wm. B. Cottrell†

Abstract: *About 4:00 a.m. on Mar. 28, 1979, Unit 2 at the Three Mile Island Nuclear Power Station experienced a turbine trip. The subsequent sequence of events involving human errors, design deficiencies, and equipment failures resulted in an accident unique in reactor operating experience to date. Although no one was injured by this incident, it has resulted in increased concern for the nuclear option and has prompted numerous investigations. This preliminary report summarizes the status of the plant and related activities through April 30, primarily on the basis of information from Nuclear Regulatory Commission (NRC) press releases, Preliminary Notification of Occurrence memorandums, and Inspection and Enforcement bulletins. No conclusions are drawn at this time, but the incident will be fully covered in a subsequent article when the findings from some of the more substantive investigating committees become available.*

On March 28, 1979, Unit 2 of the Three Mile Island Nuclear Power Station experienced a loss-of-coolant accident (LOCA) which was the result of a unique combination of equipment failures, design deficiencies, and operator errors. The initial transient was followed by several days of concern for a gas bubble trapped in the top head of the reactor pressure vessel, and, even after that problem was alleviated, not until April 28 did the plant attain the "cold shutdown" condition (a low temperature, low pressure condition in which the residual decay heat in the reactor can be removed by any of several options—in this instance by natural convection). This article summarizes events through the end of April but makes no attempt to analyze the events that occurred, principally as reported by the Nuclear Regulatory Commission (NRC). The incident is being thoroughly investigated and analyzed by several groups, and, after their work is completed, *Nuclear Safety* will carry a final report on the incident.

The Three Mile Island nuclear power station[1] is located on Three Mile Island, a 625-acre island in the Susquehanna River approximately 10 miles southeast of Harrisburg, Pa. There are two pressurized-water reactors (PWRs), both of which were designed by the Babcock & Wilcox Company, in commercial operation at the site. Unit 2 has a design electric rating of 906 MW for which the design thermal rating is 2772 MW. The circulating water system of both reactors is a closed system employing cooling towers, with only makeup water taken from the river. The two hyperbolic cooling towers required for each reactor dominate the scene in the many pictures that have been taken of the site. At the time of the incident, Unit 2 was operating at 98% of full power after being placed in commercial operation in January 1979.

INITIAL INCIDENT CHRONOLOGY

The preliminary chronology of the initial stages (first 16 h) of the incident, as reported by NRC on Apr. 5, 1979 (Ref. 2), is presented below. It should be anticipated that this chronology may be modified as a result of ongoing investigations, but, as it stands, this chronology reconstructs the essential events during the first 16 h of the incident until core cooling was restored. The preliminary notification for that date[3] is much less informative but did indicate some concern for a gas bubble in one loop.

Time‡ (approx.)	Event
About 4:00 a.m. (t = 0)	Loss of condensate pump; loss of feedwater; turbine trip
t = 3–6 s	Electromatic relief valve opens (2255 psi) to relieve pressure in reactor coolant system (RCS)
t = 9–12 s	Reactor trip on high RCS pressure (2355 psi)
t = 12–15 s	RCS pressure decays to 2205 psi (relief valve should have closed)
t = 15 s	RCS hot-leg temperature peaks at 611°F, 2147 psi (450 psi over saturation)
t = 30 s	All three auxiliary feedwater pumps running at pressure (pumps 2A and 2B started at

*W. R. Casto, Editor for the Operating Experience Section of *Nuclear Safety*, is a section head in the Operations Division of ORNL.

†W. B. Cottrell, Editor of *Nuclear Safety*, is a section head in the Engineering Technology Division of ORNL.

‡The time given is the time from the initiation of the incident, which was approximately 4:00 a.m., Mar. 28, 1979.

	turbine trip); no flow was injected since discharge valves were closed
t = 1 min	Pressurizer level indication begins to rise rapidly; steam generators A and B secondary level very low, drying out over next couple of minutes
t = 2 min	Emergency core-cooling system (ECCS) high pressure injection (HPI) initiated at 1600 psi
t = 4–11 min	Pressurizer level off scale (high); one HPI pump manually tripped at about 4 min, 30 s; second pump tripped at about 10 min, 30 s
t = 6 min	RCS flashes as pressure bottoms out at 1350 psig; hot-leg temperature of 584°F
t = 7 min, 30 s	Reactor building sump pump came on
t = 8 min	Auxiliary feedwater flow is initiated by opening closed valves
t = 8 min, 18 s	Steam-generator B pressure reached minimum
t = 8 min, 21 s	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	Reactor coolant drain/quench tank rupture disk blows at 190 psig (set point 200 psig) owing to continued discharge of electromagnetic relief valve
t = 20–60 min	System parameters stabilized in saturated condition at about 1015 psig and about 550°F
t = 1 h, 15 min	Operator trips reactor coolant (RC) pumps in loop B
t = 1 h, 40 min	Operator trips RC pumps in loop A
t = $1\frac{3}{4}$–2 h	Core begins heatup transient; hot-leg temperature begins to rise to 620°F (off scale within 14 min), and cold-leg temperature drops to 150°F (HPI water)
t = 2.3 h	Electromatic relief valve isolated by operator after steam-generator B is isolated to prevent leakage
t = 3 h	RCS pressure increases to 2150 psi, and electromatic relief valve opens
t = 3.25 h	RC drain tank pressure spike of 5 psig
t = 3.8 h	RC drain tank pressure spike of 11 psi, RCS pressure 1750; containment pressure increases from 1 to 3 psig
t = 5 h	Peak containment pressure of 4.5 psig
t = 5–6 h	RCS pressure increases from 1250 to 2100 psi
t = 7.5 h	Operator opens electromatic relief valve to depressurize RCS to attempt initiation of residual heat removal (RHR) at 400 psi
t = 8–9 h	RCS pressure decreases to about 500 psi; core flood tanks partially discharge
t = 10 h	Containment pressure spike of 28 psig; containment sprays initiated and stopped after 500 gal of NaOH injected (about 2 min of operation)
t = 13.5 h	Electromatic relief valve closed to repressurize RCS, collapse voids, and start RC pump
t = 13.5–16 h	RCS pressure increases from 650 to 2300 psi
t = 16 h	RC pump in loop A started; hot-leg temperature decreases to 560°F, and cold-leg temperature increases to 400°F, indicating flow through steam generator
After 16 hr	Steam-generator A steaming to condenser; condenser vacuum reestablished; RCS cooled to about 280°F, 1000 psi
April 4	High radiation in containment; all core thermocouples less than 460°F; using pressurizer vent valve with small makeup flow; slow cooldown; reactor building pressure negative

CHRONOLOGY OF THE SYSTEM DEGASIFICATION

The events of the preceding chronology left the reactor with one (of two operable) reactor coolant pump running and circulating reactor coolant water through steam-generator A. However, the reactor primary system was still at a relatively high pressure (slightly over 1000 psig) and temperature (core outlet temperatures) between 500 and 600°F. It is normally desirable to reduce both the temperature and pressure of the reactor so that the decay energy can be removed by the Residual (or Decay) Heat Removal System, which requires the primary system to be in a lower energy condition (i.e., less than 180 psig). However, the discovery on Thursday, March 29, that there was a significant quantity of noncondensable gases in the system precluded an immediate depressurization. It was not until mid-April that most of the noncondensable gases (including hydrogen from water–fuel cladding reactions) could be dissolved in the primary system water from which it was subsequently sparged so that the system depressurization could be initiated. Some of the significant events taken from the NRC Preliminary Notification of Event or Unusual Occurrence reports (which were issued almost daily through the month of April) are tabulated below.

March 29, 11:00 p.m.[4] System operating stably using one RC pump through steam-generator A. Pressure of ~1000 psig, and pressurizer temperature of ~550°F. Operation at these conditions is expected to continue for 24 h while the significance of gas bubbles is evaluated.

March 30, a.m.[4] Intermittent uncontrolled releases of radioactivity into the atmosphere from the contaminated RC water which, after blowing down and collecting in the containment sump, had been automatically transferred to the auxiliary building. At 11:30 a.m., NRC Chairman Hendrie suggested to Governor Thornburg that pregnant women and preschool children be evacuated from an area within 5 miles of the plant site. The highest measured off-site dose rates were a few milliroentgens per hour.

March 30, p.m.[5] Evaluation of several options to reach a "final safe state" for the fuel in view of the existence of the gas bubble; reactor being maintained in the same stable condition. Volume of the gas bubble is estimated to be approximately 1000 to 1500 ft^3 at 1000 psi. Off-site ground-level surveys in the Middletown and Goldsboro areas gave values ranging from 0.01 to 1.0 mr/h.

March 31, a.m.[6] Reactor system operation continuing unchanged with temperature slowly decreasing. The hydrogen recombiner in the containment building is operable, and additional shielding was provided.

March 31, p.m.[7] Continuing acquisition and installation of lead shielding on hydrogen recombiner. Reactor system continues stable operation. Volume of gas bubble is now estimated at 620 to 880 ft^3 at 875 psig.

April 1 (Ref. 8). No substantial change in primary system temperature or pressure. Action is taken to vent radioactive gases from the waste gas decay tanks. Shielding on hydrogen recombiners is to be completed April 2. Assessment of gas bubble continues.

April 2 (Ref. 9). Reactor pressure still being held around 1000 psi, and all fuel-element exit temperatures now below 475°F with coolant temperature of 280°F and negligible core ΔT. There was a consensus among experts that oxygen in the gas bubble was much less than originally estimated (no explosion potential); also volume of gas bubble is further reduced.

April 3 (Ref. 10). Reactor pressure remaining near 1000 psi, with bulk core coolant inlet and outlet temperatures at 280°F. Core thermocouple readings are relatively unchanged and indicate a maximum temperature of 477°F, which is well below saturation temperature for this pressure. (Only three thermocouples read above 400°F.) The gas bubble still appears to be present at a much reduced volume, with bubble size calculations still being evaluated. Degasification continues. Containment atmosphere measurements indicate about 1.9% hydrogen. One hydrogen recombiner is operating, and a 12-day time period is projected for reduction of the hydrogen concentration to about 1%.

April 4 (Ref. 11). Little change from the conditions reported on April 3. Vent valves on pressurizer have been closed, and degasification continues through the letdown system. Summary of environmental monitoring to date is as follows: water, 130 off-site water samples show no detectable radioiodine; air, 152 off-site samples indicate maximum activity detected is one-fourth of the maximum permissible concentration (MPC); milk, some radioiodine detected in some of the milk samples but the level is a factor of 300 below HEW "action" levels.

April 5–April 6 (Refs. 12, 13). Reactor conditions little changed from the report of April 3, but maximum core temperature now only 448°F (vs. 472°F). Gas from waste gas decay tanks are vented to containment.

April 7 (Ref. 14). Reactor pressure of about 1075 psi, with bulk core coolant inlet and outlet temperatures at about 285°F. At approximately 1:25 p.m. on April 6, RC pump 1A tripped, and RC pump 2A was started within about 2 min. After the change in operating pumps, there was a shift in the core thermocouple readings. The three thermocouples that had readings above 400°F are presently reading between 285 and 315°F. The central thermocouple (position 8H) reading changed from approximately 375 to 455°F and is now reading 453°F, the only reading above 400°F. The average temperature of the 30 thermocouples being monitored is 304°F.

April 7–April 8 (Ref. 15). Bulk coolant inlet and outlet temperatures of about 281°F. The average core thermocouple temperature is about 299°F, and the higher thermocouple reading (8H) is about 442°F.

At approximately 7:55 p.m. on April 7, the licensee began lowering RCS pressure in 50-psi increments at a maximum rate of 5 psi/min. This will continue until the pressure reaches 500 psi, which will provide a 100-psi safety margin above saturation for the current temperature of the highest reading thermocouple. This is a step toward cold shutdown and includes degasification to prevent bubble formation as pressure and temperatures decrease.

During the initial pressure decrease to 700 psi, the auxiliary building stack monitors showed an increase of a factor of 10 at 10:13 p.m. on April 7. Later information indicates that about 1400 gal of borated water was added to the makeup tank during the initial pressure reduction and caused some gas to leak from the vent header. The ARMS helicopter reported a slight increase in readings downwind (south) of the site. Pressurė was held steady for a short period, and the auxiliary stack monitors decreased to the original readings. During the following pressure cycles, there have been no increases in the radiation readings.

April 9 (Ref. 16). Bulk coolant inlet and outlet temperatures of about 280°F. The average core thermocouple temperature is about 300°F, and the highest thermocouple reading (8H) is about 425°F.

At approximately 4:30 a.m., the RCS pressure reached the 400-psig endpoint established for the second degassing evolution. At lower pressures in the 400- to 1000-psig range, noise monitoring indicated possible presence of some gas in loop B of the reactor cooling system. Noise monitoring verified re-solution of gas with time. The operating RC pump vibration increased to 8.5 to 9 mils, but the level of vibration was still significantly below the limit (30 mils). Pressure variation for degassing is continuing. After reduction to 400 psig, the licensee plans to increase pressure to the 900- to 1000-psig range, and a phased cooldown is under consideration as the next step.

The licensee requested and received permission to temporarily change the minimum pressurizer level to 150 in. from 200 in. to prevent high pressurizer levels on pressure decreases.

At approximately 1:20 a.m. on April 8, the RCS began to heat up owing to a decrease in steam-generator level. Steam-generator A level was increased to decrease the primary temperature.

April 9–April 10 (Ref. 17). Bulk coolant inlet and outlet temperatures remaining at about 280°F. The average core thermocouple temperature is about 295°F, and the highest thermocouple reading (8H) is about 400°F.

A 24-h period of additional degasification by reducing primary pressure to 400 psig in small decrements was completed on April 9. No significant change in RC pump vibration occurred during this period. Noise measurements did indicate some gas in the coolant at lower pressures, with return into solution with time. The licensee plans to repeat the degassing operation, cycling down to approximately 300 psig, and subsequently to hold reactor coolant system pressure at approximately 1000 psig. A phased cooldown is under consideration as the next step.

April 11 (Ref. 18). Bulk coolant inlet and outlet temperatures remaining at approximately 280°F. The peak core thermocouples have declined to less than 400°F for the first time; the highest thermocouple reading is 398°F.

Degassing operations were continued; however, after cycling down to 425 psig, pressure had to be increased because the system letdown flow rate was not sufficient to prevent an increase in the pressurizer level caused by normal coolant pump seal water leakage into the RCS. Pressure was increased to 550 psig where some degassing occurred. Pressure was subsequently increased to approximately 940 psig, where it is being held while the pressurizer level is being reduced. Continued degassing operations, with reactor pressure reduced to 300 psig, is being reexamined.

April 12 (Ref. 19). Bulk coolant inlet and outlet temperatures remaining at approximately 280°F. The peak core thermocouples remain less than 400°F, except for one thermocouple that read 401°F during reduced pressure operation.

The degassing operations were completed at about 1:15 a.m. The minimum RCS pressure was 303 psig.

Noise analysis evaluations indicate considerable degassing took place during these operations. Pressure is being returned to about 1000 psig and will be held at that level.

April 13 (Ref. 20). Bulk coolant inlet and outlet temperatures remaining at approximately 280°F. The peak core thermocouple readings have declined to below 385°F. The primary system pressure is being maintained between 950 and 1000 psig.

There are presently two of the three original pressurizer level channels in operation. The pressurizer level indicator that was reported to have failed on April 11 started to function again at 7:55 p.m. on April 12 and has been tracking reasonably well.

The hydrogen recombiner tripped off at 1:15 a.m. (burned-out heaters). The hydrogen concentration in the containment building was about 1.5% at 10:00 p.m. on April 12. A decision has not been made whether to replace the heaters or to initiate operation of the backup recombiner.

April 13–April 28 (Refs. 21, 22). At 10:03 a.m., cooldown of the primary coolant system was initiated marking the first step toward placing the reactor into natural circulation. It is anticipated that the primary system would be cooled from 280°F to approximately 230°F during this phase. As of 2:00 a.m. on April 14, the primary coolant temperature had decreased to approximately 250°F, and cooldown had slowed considerably. Four of the incore thermocouple readings remained above 300°F, with the highest at 350°F.

The natural circulation mode was finally attained on April 28—somewhat ahead of schedule—because of the degradation of the level indication in the pressurizer.,

PRELIMINARY NRC ASSESSMENT OF CAUSES

On April 5 the NRC released a bulletin[2] in which it identified the following six potential human, design, and mechanical failures that contributed to the incident:

1. At the time of the initiating event, there was loss of feedwater, with both of the auxiliary feedwater trains 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 pressurizer level indication may have lead to erroneous inferences of high level in the RCS. The pressurizer level indication apparently led the operators to prematurely terminate HPI flow, even though substantial voids existed in the RCS.

4. Because the containment does not isolate on 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. The water entered the radioactive waste-treatment system in the auxiliary building, where some of it overflowed to the floor. Outgassing from the water and discharge through the auxiliary building ventilation system and filters was the principal source of the off-site release of radioactive noble gases.

5. Subsequently the HPI system was intermittently operated in an attempt to control primary coolant inventory losses through the electromatic relief valve, which were apparently based on the pressurizer level indication. Owing to the presence of steam or noncondensable voids elsewhere in the RCS, this led to a further reduction in primary coolant inventory.

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

NRC INSTRUCTIONS TO OTHER REACTOR LICENSEES

As the Three Mile Island incident evolved and was subsequently contained, the NRC issued a series of Inspection and Enforcement bulletins to other reactor licensees [in addition to the Preliminary Notification of Occurrences (PNOs) that were issued almost daily from the start of the accident through the end of April]. These Inspection and Enforcement bulletins contained instructions to the various reactor licensee holders regarding certain actions they were directed to take in view of the experience at Three Mile Island. The first such bulletin on Apr. 1, 1979, was sent to all Babcock & Wilcox (B&W) pressurized-water-reactor facilities, and subsequent bulletins encompassed all licensed water-reactor facilities. Several of these bulletins and excerpts from some of them are presented below:

Apr. 1, 1979, to B&W Facilities[23]

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 & 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.

Apr. 5, 1979, to B&W Licensees

This bulletin clarified and expanded on IE Bulletin No. 79-05 (Ref. 24).

Apr. 11, 1979, to Combustion Engineering and Westinghouse Licensees[25]

1. Review the description of circumstances described in Enclosure 1 of IE Bulletin 79-05 and the preliminary chronology of the TMI-2 3/28/79 accident included in Enclosure 1 to IE Bulletin 79-05A.
   a. This review should be directed toward understanding: (1) 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; (2) the apparent operational errors which led to the eventual core damage; and (3) the necessity to systematically analyze plant conditions and parameters and take appropriate corrective action.
   b. Operations personnel should be instructed to: (1) not override automatic action or engineered safety features without careful review of plant conditions; and (2) not make operational decisions based on a single plant parameter indication when a confirmatory indication is available.
   c. All licensed operators and plant management and supervision with operational responsibilities shall participate in this review and such participation shall be documented in plant records.
2. For pressurized water reactor facilities, 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.
3. For pressurized water reactor facilities that use pressurizer water level coincident with pressurizer pressure for automatic initiation of safety injection into the reactor coolant system, instruct operators to manually initiate safety injection when the pressurizer pressure indication reaches the actuation set point whether or

not the level indication has dropped to the actuation set point.

4. 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.
5. For pressurized water reactor facilities for which the auxiliary feedwater system is not automatically initiated, prepare and implement immediately procedures which require the stationing of an individual (with no other assigned concurrent duties and in direct and continuous communication with the control room) to promptly initiate auxiliary feedwater to the steam generator(s) for those transients or accidents the consequences of which can be limited by such action.
6. For all pressurized water reactors, prepare and implement immediately procedures which:
   a. Identify those plant indications (such as valve discharge piping temperature, valve position indication, or valve discharge relief tank temperature or pressure indication) which plant operators may utilize to determine that pressurizer power operated relief valve(s) are open, and
   b. Direct the plant operators to manually close the power operated relief block valve(s) when reactor coolant system pressure is reduced to the set point for normal automatic closure of the power operated relief valve(s) and the valve(s) fail to close.
7. Review the action directed by the operating procedures and training instructions to ensure that:
   a. Operators do not override automatic actions of engineered safety features without careful review of plant condtions.
   b. Operators are provided additional information and instructions to not rely upon any one plant parameter but to also examine other related indications in evaluating plant conditions.
8. Review all safety-related valve positions, positioning requirements and positive controls to assure that valves remain 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, testing, plant and system startup, and supervisory periodic (daily/shift checks, etc.) surveillance to ensure that such valves are returned to their correct positions following necessary manipulations and are maintained in their proper positions during all operational modes.
9. 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, venting or other release 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.
   c. The basis on which continued operability of the above features is assured.
10. Review and modify as necessary your maintenance and test procedures to ensure that they require:
    a. Verification, by test or inspection per technical specifications, 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. Explicit notification of involved reactor operating personnel whenever a safety-related system is removed from and returned to service.
11. Review your prompt reporting procedures for NRC notification to assure very early notification of serious events.

Apr. 14, 1979, to General Electric Licensees

This bulletin was similar to IE Bulletin No. 79-06 but was oriented specifically to the General Electric boiling-water reactors.[26]

Apr. 18, 1979, to Westinghouse Licensees

This bulletin clarified and expanded on IE Bulletin No. 79-06, as applied specifically to Westinghouse nuclear plants.[27]

Apr. 14, 1979, to Combustion Engineering Licensees

This bulletin clarified and expanded on IE Bulletin No. 79-06, as applied specifically to Combustion Engineering nuclear plants.[28]

Apr. 21, 1979, to B&W Licensees

This bulletin supersedes and contains additional requirements over those in IE Bulletin No. 79-05A (Ref. 29).

INVESTIGATING COMMITTEES

Although the public consequences (as measured in health effects) were slight—no deaths and the maximum exposure was less than 80 mr—the public alarm and attention that was directed toward the Three Mile Island incident has inevitably led to the establishment of a number of "investigating" committees. At this writing the authors are aware of the following committees, but there is no assurance that this list is complete.

1. Presidential Commission.
2. Nuclear Regulatory Commission.
3. Electric Power Research Institute.
4. Advisory Committee on Reactor Safeguards.
5. Metropolitan Edison Company.
6. Babcock & Wilcox Company.
7. Energy Subcommittee of the House Interior Committee.
8. Health Committee of the Senate Human Resources Committee.
9. Energy Subcommittee of the Joint Economic Committee.
10. Environmental Protection Agency.
11. Energy and Nuclear Proliferation Subcommittee of the Senate Governmental Affairs Committee.
12. Energy Subcommittee of the House Government Operations Committee.
13. Energy and Power Subcommittee of the House Commerce Committee.
14. General Accounting Office.

One cannot help but question whether all these investigations are necessary.

INTERLUDE

As indicated in the introduction to this article, this is a preliminary report. A final report on the Three Mile Island incident will be published in *Nuclear Safety* when the more substantiative of the investigations have been completed so that a complete picture can be presented.

REFERENCES

1. F. A. Heddleson, ***Design Data and Safety Features of Commercial Nuclear Power Plants***, AEC Report ORNL-NSIC-55, Vol. II, pp. 85-90, Oak Ridge National Laboratory, 1972.
2. NRC Inspection and Enforcement Bulletin No. 79-05A, Apr. 5, 1979.
3. NRC Preliminary Notification No. PN-NRC: I-38A, Mar. 28, 1979.
4. NRC Preliminary Notification No. PNO-79-67B, Mar. 30, 1979.
5. NRC Preliminary Notification No. PNO-79-67C, Mar. 30, 1979.
6. NRC Preliminary Notification No. PNO-79-67E, Mar. 31, 1979.
7. NRC Preliminary Notification No. PNO-79-67F, Mar. 31, 1979.
8. NRC Preliminary Notification No. PNO-79-67G, Apr. 1, 1979.
9. NRC Preliminary Notification No. PNO-79-67H, Apr. 2, 1979.
10. NRC Preliminary Notification No. PNO-79-67I, Apr. 3, 1979.
11. NRC Preliminary Notification No. PNO-79-67J, Apr. 4, 1979.
12. NRC Preliminary Notification No. PNO-79-67K, Apr. 5, 1979.
13. NRC Preliminary Notification No. PNO-79-67L, Apr. 6, 1979.
14. NRC Preliminary Notification No. PNO-79-67M, Apr. 7, 1979.
15. NRC Preliminary Notification No. PNO-79-67N, Apr. 8, 1979.
16. NRC Preliminary Notification No. PNO-79-67P, Apr. 9, 1979.
17. NRC Preliminary Notification No. PNO-79-67Q, Apr. 10, 1979.
18. NRC Preliminary Notification No. PNO-79-67R, Apr. 11, 1979.
19. NRC Preliminary Notification No. PNO-79-67S, Apr. 12, 1979.
20. NRC Preliminary Notification No. PNO-79-67T, Apr. 13, 1979.
21. NRC Preliminary Notification No. PNO-79-67U, Apr. 14, 1979.
22. NRC Preliminary Notification No. PNO-79-67AI, Apr. 28, 1979.
23. NRC Inspection and Enforcement Bulletin No. 79-05, Apr. 1, 1979.
24. NRC Inspection and Enforcement Bulletin No. 79-05A, Apr. 5, 1979.
25. NRC Inspection and Enforcement Bulletin No. 79-06, Apr. 11, 1979.
26. NRC Inspection and Enforcement Bulletin No. 79-08, Apr. 14, 1979.
27. NRC Inspection and Enforcement Bulletin No. 79-06A, Apr. 18, 1979.
28. NRC Inspection and Enforcement Bulletin No. 79-06B, Apr. 14, 1979.
29. NRC Inspection and Enforcement Bulletin No. 79-05B, Apr. 21, 1979.

Developments Pertaining to the Three Mile Island Accident

By Wm. B. Cottrell

A preliminary report on the Three Mile Island accident of March 28, 1979, was included in the previous issue of *Nuclear Safety*.[1] As was stated in that article, a final report on the accident will be presented in *Nuclear Safety* when the various investigating committees report on their findings. Most of these reports should be available by the end of the year. However, some of the developments of the past 2 months pertaining to the accident are of general interest and will be summarized here. No attempt is made here to present a comprehensive review of the accident nor even to evaluate the material that has become available; rather, given the interest in the subject, this article will merely call attention to the available information. (In addition, the report by the Ad Hoc Dose Assessment Group is summarized in the previous section of this issue of *Nuclear Safety*.) The developments reported here fall into the following topics: Lessons Learned Task Force, Advisory Committtee on Reactor Safeguards (ACRS) Reports on Three Mile Island, Congressional Investigations, Metropolitan Edison Company Interim Report, Nuclear Regulatory Commission (NRC) Report on Babcock & Wilcox (B&W) Feedwater Transients, Tennessee Valley Authority (TVA) Nuclear Program Review, Radioactivity Sampling, Liability Insurance Payments, and a catchall heading entitled Miscellaneous Actions of Note.

LESSONS LEARNED TASK FORCE

In May, Harold R. Denton, Director of the NRC Office of Nuclear Reactor Regulation, announced the establishment of a "Lessons Learned Task Force" that would have the responsibility to sort through the areas of possible regulatory improvements which might then be implemented by changes in legislation, regulations, staff positions, review procedures, or otherwise. In view of the time needed to complete this evaluation (including review and revision), Denton anticipates an effective 3-month moratorium on the issuance of new licenses. The areas of immediate interest to this task force include:

1. Reactor operator training and licensing.
2. Reactor transient and accident analysis.
3. Licensing requirements for safety and process equipment, instrumentation, and controls.
4. Off-site and on-site emergency preparations and procedures.
5. Reactor siting.
6. Licensee technical qualification.
7. Nuclear Reactor Regulation (NRR) accident response role, capability, and management.
8. Reactor operating experience.
9. Environmental effects.
10. Licensing requirements for postaccident monitoring and controls.
11. Postaccident cleanup and recovery.
12. NRR engineering evaluation of the event sequence at Three Mile Island Unit 2.

In the meantime, the NRC Office of Inspection and Enforcement is continuing to inform other nuclear utilities of the status of Three Mile Island, to request additional safety assessments, and to review their operator training programs.

ACRS REPORTS ON THREE MILE ISLAND

On Apr. 17, 1979, recommendations pertaining to the Three Mile Island accident were made orally to the NRC by the ACRS. Subsequently the recommendations were reviewed by all ACRS members during its meeting on May 10–12 and submitted in a report on May 16 in a somewhat amplified form. The essential contents of that report and of another report of the same date containing additional recommendations are presented below.

Interim Report 2 (Ref. 3)

Natural Circulation: Procedures

It is evident from the experience at TMI-2 [Three Mile Island Unit 2] that there was failure to establish natural circulation of water in the primary system and failure to recognize in a timely manner that natural circulation had not been achieved. The need for natural circulation under certain circumstances is common to all PWRs [pressurized-water reactors].

The Committee recommends that procedures be developed by all operators of PWRs for initiating natural circulation in a safe manner and for providing the operator with assurance that circulation has in fact been established. These procedures should take into account the behavior of the systems under a variety of abnormal conditions.

As a first step, the NRC Staff should initiate immediately a survey of operating procedures for achieving natural circulation, including the case when offsite power is lost. At the same time, the operators of all PWR plants should be requested to develop detailed analyses of the behavior of their plants following anticipated transients and small breaks in the primary system, with appropriate consideration of potential abnormal conditions, operator errors and failures of equipment, power sources, or instrumentation. These analyses are necessary for the development of suitable operating procedures. The review and evaluation of these analyses by the NRC Staff should receive a priority consistent with the priority being given to changes in operating procedures.

Natural Circulation: Pressurizer Heaters

The use of natural circulation for decay heat removal following an accident in a PWR normally requires the maintenance of a suitable overpressure on the reactor coolant system in order to prevent the generation of steam which can impede circulation. For many transients, maintenance of this overpressure is best accomplished by use of the pressurizer heaters.

Although the pressurizer heaters at TMI-2 continued to receive power from offsite sources during the entire accident, the availability of offsite power cannot be assured for all transients or accidents during which, or following which, natural circulation must be established. The Committee recommends that the NRC Staff initiate immediately a survey of all PWRs licensed for operation to determine whether the pressurizer heaters are now or can be supplied with power from qualified onsite sources with suitable redundancy.

Natural Circulation: Saturation Conditions

The plant operators should be informed adequately at all times of those conditions in the reactor coolant system that might affect their capability to place the system in the natural circulation mode or to sustain it in such a mode. Information indicating that coolant pressure is approaching the saturation pressure corresponding to the core exit temperature would be especially useful, since an impending loss of overpressure would signal to the operator a potential loss of natural circulation. This information can be derived from available pressurizer pressure and hot leg temperature measurements, in conjunction with conventional steam tables.

The Committee recommends that information for detecting an approach to saturation pressure be displayed to the operator in a suitable form at all times. Since there may be several equally acceptable means of providing this information, there is no need for the NRC Staff to assign a high priority to the development of prescriptive requirements for such displays. However, a reasonably early request that licensees and vendors consider and comment on the need for such a display would be appropriate.

Core Exit Thermocouples

The NRC Staff should request licensees and vendors to consider whether the core exit temperature measurements might be utilized, where available, to provide additional indication regarding natural circulation or the status of the core. For the latter purpose, it is recommended that the full temperature range of the core exit thermocouples be utilized. At TMI-2, the temperatures displayed and recorded did not include the full range of the thermocouples.

The Committee believes it would be appropriate for the NRC Staff to request licensees and vendors to consider and comment on this recommendation. This request should be made as soon as convenient and the time allowed for responses should be such as not to degrade responses on higher priority matters. Plant changes that might result eventually from consideration of this recommendation would not at this time seem to require a high priority.

Instrumentation to Follow the Course of an Accident

The ability to follow and predict the course of an accident is essential for its mitigation and for the provision of credible and reliable predictions of potential offsite consequences. Instrumentation to follow the course of an accident in power reactors of all types has long been a concern of the ACRS, is the subject of Regulatory Guide 1.97 (which has not yet been implemented on an operating plant), and is the subject of an NRC Staff Task Action Plan for the resolution of generic issues.

The Committee believes that the positions of Regulatory Guide 1.97 should be reviewed, and redefined as necessary, and that the Task Action Plan should be reexamined, as soon as manpower is available. The lessons learned from TMI-2 should be the bases for these reviews. For example, improved sampling procedures under accident conditions should be considered.

Although review and reexamination of existing criteria may take some time, the studies completed to date, together with the understanding gained from the accident at TMI-2, should provide sufficient basis for planned and appropriately phased actions. The Committee believes that the installation of improved instrumentation on operating reactors of all types should be under way within one year.

Reactor Safety Research

The ACRS recommends that safety research on the behavior of light-water reactors during anomalous transients be initiated as soon as possible and be assigned a high priority. The ACRS would expect to see plans and proposals within about three months, preliminary results within an additional six months, and more comprehensive results within a year.

Of particular interest would be the development of the capability to simulate a wide range of postulated transient or accident conditions, including various abnormal or low probability mechanical failures, electrical failures, or human errors, in order to gain increased insight into measures that can be taken to improve safety.

The new program of research to improve reactor safety has been initiated only recently, and then only on a relatively small scale. The Committee reiterates its previous recommendations that this program be pursued and its expansion sought by the Commission with a greater sense of urgency.

Status Monitoring

Although the closed auxiliary feedwater system valves may not have contributed directly or significantly to the core damage or environmental releases at TMI-2, the potentially much more severe consequences of unavailability of engineered safety features in plants of any type is of concern and deserving of attention. Status monitoring not dependent chiefly on administrative control, and thus possibly less subject to human error, might help assure the availability of essential features.

A request should be made within the next few months that licensees consider additional status monitoring of various engineered safety features and their supporting services. The NRC Staff should begin studies on the advantages and disadvantages of such monitoring on about the same time scale. Responses from licensees should be expected in about one year, at which time the NRC Staff should be in a position to review and evaluate them.

The Committee recognizes that some of the recommended actions in this report have already been taken by the NRC Staff.

Interim Report 3 (Ref. 4)

Reactor Pressure Vessel Level Indication

The Committee believes that it would be prudent to consider expeditiously the provision of instrumentation that will provide an unambiguous indication of the level of fluid in the reactor vessel. We suggest that licensees of all pressurized water reactors be requested to submit design proposals and schedules for accomplishing this action. This would assure the timely availability of reviewed designs if the Staff ongoing studies should indicate that early implementation is required. The Committee believes that as a minimum, the level indication should range from the bottom of the hot leg piping to the reactor vessel flange area.

Operator Training and Qualification

The NRC Staff should examine operator qualifications, training, and licensing to determine what changes are needed. Consideration should be given to educational background, to training methods, and to content of the training program. Attention should also be given to testing methods, with specific concern for the ability of the testing methods to predict operator capability. Examination of licensing procedures should determine whether they are responsive to new information that is developed about plant or operator performance. Effort should also be made to determine whether results of examinations can be correlated with operator ability. Requalification training and testing should be similarly examined to insure that they take account of information that is developed by operation in the plant, and to determine that relevant information about other plants is made available to operators, and is made part of the training and requalification program. As part of this and of other more extensive studies, continuing attention must be given to the amount of information which an operator can assimilate and use in normal and in emergency situations and to the best method of presenting the information to the operator. The use and limitations of simulators for operator training should receive careful consideration.

Evaluation of Licensee Event Reports

Because of the potentially valuable information contained in Licensee Event Reports (LERs), the Committee recommends that the NRC Staff establish formal procedures for the use of this information in the training of supervisory and maintenance staffs and in the licensing and requalification of operating personnel at commercial nuclear power plants. The information in LERs may also be useful in anticipating safety problems. At the present time some utilities routinely request that they be provided copies of all LERs applicable to plants of the type they operate or to specific systems and components in a given class of plants similar to their plant. Certain reactor vendors have made similar requests and use the LERs to review and evaluate the performance of their plants. In addition, the NRC operator licensing staff has indicated that they use LERs in reviewing operating experience at commercial facilities.

The large number of LERs that attribute the cause to personnel error would tend to indicate that a formalized program of LER review would be useful in the training, licensing and requalification of nuclear power plant personnel. The extent to which such a program could be used to anticipate safety problems should also be considered.

Operating Procedures

Safety aspects of individual reactors during normal operation and under accident conditions are reviewed in detail by the NRC Staff and discussed with the ACRS. Acceptable limits for normal operation are formalized by Technical Specifications, submitted by the licensee and approved by the NRC Staff. Operating procedures for severe transients have received less detailed review by the NRC Staff. It appears that such procedures would benefit from review by an interdisciplinary team which includes personnel expert both in operations and in system behavior. Also, for the longer term, there may be merit in considering the development of more standardized formats for such procedures.

Reliability of Electric Power Supplies

During the past several years, there have been several operating experiences involving a loss of a-c power to important engineered safeguards. The ACRS believes it important that a comprehensive reexamination be made by the NRC and the reactor licensees of the adequacy of design, testing, and maintenance of offsite and onsite a-c and d-c power supplies. In particular, failure modes and effects analyses should be made, if not already performed, more systematic testing of power system reliability, including abnormal or anomalous system transients, should be considered, and improved quality assurance and status monitoring of power supply systems should be sought.

Analysis of Transients

The ACRS recommends that each licensee and holder of a construction permit be asked to make a detailed evaluation of his current capability to withstand station blackout (loss of offsite and onsite a-c power) including additional complicating factors that might be reasonably considered. The evaluation should include examination of natural circulation capability, the continuing availability of components needed for long-term cooling, and the potential for improvement in capability to survive extended station blackout.

The ACRS also recommends that each licensee and construction permit holder should examine a wide range of anomalous transients and degraded accident conditions which might lead to water hammer. Methods of controlling or preventing such conditions should be evaluated, as should research to provide a better basis for such evaluations. The Committee expects it would be appropriate to have such studies done generically first, for classes of reactor designs and system types.

Emergency Planning

An effort should be undertaken to plan and define the role NRC will play in emergencies and what their contribution and interaction will be with the licensee and other emergency plan participants including other government agencies, industry representatives, and national laboratories. Such planning should consider:

—assurance that formal documentation of plans, procedures and organization are in place for action in an emergency,
—designation of a technical advisory team with names and alternates for the anticipated needs of an emergency situation,
—compilation of an inventory of equipment and materials which may be needed for unusual conditions including its description, location, availability and the organization which controls its release.

The Committee recommends that each licensee be asked to review and revise within about three months:
—his bases for obtaining offsite advice and assistance in emergencies, from within and outside the company,
—current bases for notifying and providing information to authorities offsite in case of emergency.

This review and evaluation should be in terms of accidents having a broad range of consequences. The results of this review should be reported to the NRC.

Decontamination and Recovery

The Committee wishes to call attention to the importance of a program designed to learn directly about the behavior, failure modes, survivability, and other aspects of component and system behavior at TMI-2 as part of the long-term recovery process. This program should also examine the lessons learned at TMI-2 to determine if design changes are necessary to facilitate the decontamination and recovery of major nuclear power plant systems.

Safety Review Procedures

The TMI-2 accident has imposed large new pressures on the availability of manpower resources within the NRC Staff. If progress is to be expedited on the new questions which have arisen and on existing unresolved safety issues, the ACRS believes that new mechanisms should be sought and implemented. For those safety concerns where such a mechanism is appropriate the Committee recommends that the Commission should request licensees to perform suitable studies on a timely basis, including an evaluation of the pros and cons, and proposals for possible implementation of safety improvements. The NRC Staff should concurrently establish its own capability to evaluate such studies by arranging for support by its consultants and contractors. In this fashion, the Committee anticipates that the information on which judgments will be based can be developed much more expeditiously, and an earlier resolution of many safety concerns may be achieved.

Capability of the NRC Staff

The Committee recommends that the capability of the NRC Staff to deal with basic and engineering problems in what may be termed broadly as reactor and fuel cycle chemistry be augmented expeditiously. This should include establishment of expertise within the NRC, with assistance arranged from consultants and contractors, in such important technical areas as the behavior of PWR and BWR coolants and other materials under radiation conditions; generation, handling and disposal of radiolytic or other hydrogen at nuclear facilities; performance of various chemical additives in containment sprays; processing and disposal techniques for low and high level radioactive wastes; chemical operations in other parts of the nuclear fuel cycle; and in the chemical treatment operations involved in recovery, decontamination, or decommissioning of nuclear facilities. The Committee wishes to emphasize the importance of providing this expertise in both the research and licensing management elements of the NRC.

Single Failure Criterion

The NRC should begin a study to determine if use of the single failure criterion establishes an appropriate level of reliability for reactor safety systems. Operating experience suggests that multiple failures and common mode failures are encountered with sufficient frequency that they need more specific consideration. This study should be accompanied by concurrent consideration of how the licensing process can be modified to take account of a new set of criteria as appropriate.

Safety Research

The ACRS believes that, as a result of the TMI-2 accident, various safety research areas will warrant initiation or much greater emphasis, as appropriate. The Committee suggests that consideration be given to an augmentation of the NRC safety research budget for FY 80.

Also, the Committee believes that a larger part of the safety research program should be oriented toward exploratory research as contrasted to confirmatory research, with some degree of freedom from immediate licensing requirements. The ACRS plans to have a Subcommittee meeting on this subject with representatives of the NRC Office of Nuclear Regulatory Research in the near future.

The Committee is continuing to review these matters and will report further as additional recommendations are developed.

CONGRESSIONAL INVESTIGATIONS

At least six Congressional subcommittees held hearings during May and June in which the subject matter involved some aspect of the Three Mile Island accident. These six committees and the highlights of their respective hearings are cited below:

Subcommittee on Nuclear Regulation (of Senate Committee on Environment and Public Works)[5]

This subcommittee held hearings in early May at which NRC officials briefed the committee on NRC's role at Three Mile Island. Pennsylvania Public Utility Commissioner W. Wilson stated that it seemed appropriate for the federal government to assist in the Three Mile Island recovery and replacement power costs.

Subcommittee on Military Installations (of the House Armed Services Committee)[6]

A hearing in mid-May of this committee heard testimony by Federal Civil Defense Director B. R. Tirana to the effect that Three Mile Island was a highly complex technical disaster that placed a heavy burden on emergency preparedness agencies.

Subcommittee on Energy and the Environment (of the House Committee on Interior and Insular Affairs)[7]

In a meeting in late May, this committee heard testimony from the principal Three Mile Island participants from Metropolitan Edison Company, NRC, and Babcock & Wilcox Company, as well as from its own task force which Chairman Udall had established to inquire into the Three Mile Island accident. Of interest is the task force conclusion that design error was the main contributor to the accident, whereas NRC attributes the principal contribution to operator error. Subsequent meetings[8,9] were held the last week of May and the first of June concerning a variety of issues, including nuclear waste disposal, nuclear regulation, as well as Three Mile Island.

Subcommittee on Investigations of the Civil Service Committee (of the House Subcommittee on Manpower Utilization)[10]

On June 6, 1979, this committee heard from NRC personnel regarding the adequacy of staffing and the efficiency of administration of the NRC's on-site nuclear inspection program.

Subcommittee on Environment Energy and Natural Resources (of the House Committee on Government Operations)[11,12]

This committee has held three hearings (as of May 17) regarding emergency planning for nuclear power-plant accidents. Although there appeared to be a general consensus regarding the need for better planning, there was a wide difference of opinion among federal, state, and other witnesses as to who was responsible for the present state of affairs.

Subcommittee on Energy Research and Production (of House Committee on Science and Technology)[13,14]

This committee conducted 2 weeks of hearings in mid-May regarding safety issues of nuclear power production in light of the Three Mile Island accident. In addition to Congressional testimony, the witnesses included leaders from many elements of the nuclear community. Their testimony was primarily technical and, in addition to the Three Mile Island and related safety issues, encompassed the nuclear waste-disposal issue.

METROPOLITAN EDISON INTERIM REPORT[15]

On May 15, 1979, the Metropolitan Edison Company, owner and operator of the Three Mile Island Nuclear Power Station, submitted to the NRC a preliminary report on the Mar. 28, 1979, accident. The document contains more than 100 pages and covers (1) sequence of events (for the first 24 h), (2) recovery organization, (3) plant modifications, and (4) radiological monitoring, each of which is summarized briefly.

Sequence of Events

This report provides additional detail to the Apr. 16, 1979, issue of the sequence of events of the Mar. 28, 1979, accident at Three Mile Island Unit 2. The report should still be considered as preliminary since investigation and data analysis are ongoing and continue to provide new insights into the Three Mile Island 2 accident. As new information and understanding are developed, the report will be updated.

The figures in the report represent the compilation of data from various installed instrumentation and recording sources. Future revisions of the report will identify those sources.

Annotations included along with the chronology of events, in addition to providing periodic assessments of the plant status, represent input culled from interviews with the operating staff conducted by Metropolitan Edison.

Recovery Organization

On Monday, Apr. 2, 1979, the Three Mile Island Unit 2 Recovery Organization was initiated. Owing to the constraints of the crisis, it was recognized then that the organization would be continually evaluated in light of the conditions that would exist and the tasks at hand and that refinements and modifications would take place as appropriate.

The Recovery Organization consisted of an integration of General Public Utilities personnel with senior, experienced people from other utilities and nuclear industry organizations across the country.

The Recovery Organization focused on the following priorities:

1. Maintaining the current plant operations in the safest conditions.
2. Containing the release of radioactivity to minimize exposure to the public and on-site personnel.
3. Making a reliable safe transition to a benign and reliable long-term cooling mode for the plant.
4. Reinforcing the capability of the plant to assure long-term cooling.

Plant Modifications

Several modifications to Three Mile Island 2 systems were made or considered after the accident. The modifications were undertaken to augment the existing systems for both the containment of radioactivity and the control of plant conditions during the establishment of long-term cooling.

This section includes a discussion of the various modifications that have been made. In addition, a book of drawings and diagrams pertaining to these modifications is enclosed for reference. The following systems are covered:

Hydrogen recombiners.
Auxiliary and fuel-handling building supplementary air filtration systems.
Condenser air extraction filtration system.
Fuel pool waste-storage system.
Upgraded decay-heat-removal system.
Steam-generator B closed-loop cooling system.

Portable disposable demineralizer system.
Nuclear river-water system.
Secondary services closed cooling-water system.
Steam-generator A closed-loop cooling system.
Feedwater bypass lines.
Alternate decay-heat-removal system.

Standby reactor coolant pressure control system.
Balance-of-plant electric-power system.
Liquid radioactive waste-processing system (Epicor II).
Trash compactor.
Staging facilities for dewatered resins and evaporator bottoms.
Epicor I liquid rad-waste treatment system.

Radiological Monitoring

This report uses data collected in monitoring programs operated by the Metropolitan Edison Company. At this writing, an extensive amount of data collected by others has not yet been evaluated in detail. However, the Metropolitan Edison program is comprehensive enough and sensitive enough to make an accurate assessment of radiological impact, including measured releases, estimated release rates, meteorological data, and radiological environmental monitoring program.

Available data were used to assess the radiation doses received by individuals and affected populations off-site in the period following the start of the accident. In addition, comprehensive dose analyses have been performed.

NRC REPORT ON B&W FEEDWATER TRANSIENTS[16]

On Mar. 28, 1979, the Three Mile Island Unit 2 nuclear power plant experienced a feedwater transient that, through an unusual sequence of failures, led to a small break loss-of-coolant accident and resulted in significant core damage. The failures that were experienced occurred in the general areas of design, equipment malfunction, and human error. In response to this event, a task group was formed to provide an early assessment of the generic aspects of the feedwater transient and the related ensuing events at Three Mile Island 2 to determine bases for continued safe operation of other reactor plants similar to Three Mile Island 2 that were designed by the Babcock & Wilcox Company (B&W). Consideration was given by the task group to initiating events other than loss of feedwater where it was determined that such events could lead to a similar transient. In addition, consideration was given to possible impact on other PWR plants designed by Westinghouse and Combustion Engineering.

A recent review by the staff on the frequency of feedwater transients occurring in B&W plants indicates that 27 transients have occurred in nine plants during the past year. This corresponds to a frequency of three per year per plant. The corresponding rate for the other PWR plants is about two per year per plant.

The results of this assessment are presented in this report by the task group in the form of a set of findings and recommendations in each of the principal review areas. Additional review of the accident is continuing and further information is being obtained and evaluated. Any new information will be reviewed and modifications to the results of the initial review will be made as appropriate.

Many actions have been taken since the Three Mile Island 2 event by the staff and industry to minimize the likelihood of recurrence, including the shutdown of the four operating B&W facilities for short-term corrective actions, which will also be taken on the other B&W plants before they restart. As this response is being published, there are other ongoing activities, including discussions with Westinghouse, Combustion Engineering, and various utilities, to further improve the safety margins in these plants. Thus this is a status report and is not considered to be a complete and final set of recommended actions. It is not a general critique of licensee and NRC response to the accident. Such review will follow while other ideas are being formulated, but that is beyond the scope of this report. It is likely that other actions, including long-term actions, will be required as the overall review of the Three Mile Island 2 accident progresses.

Prior to the Three Mile Island 2 accident, the general approach used for accident analyses was to ensure conservatism in the analysis models and results. Consideration has been given to the development of best-estimate codes, but licensing calculations were done on a conservative basis. It is recognized that shortcomings resulted from this approach. For example, the analysis of the Sept. 24, 1977, transient at Davis–Besse did not include the phenomenon of voiding in the core and long-term natural circulation cooling. Other areas that need to be reevaluated include the use of safety and nonsafety grade equipment for the termination of transients and mitigation of accidents.

On the basis of the results of this interim review, the task group concludes that certain design improvements and other actions already being implemented on B&W plants in accordance with NRC orders are necessary before plant operation can be resumed. These actions are being specified in the shutdown orders that resulted from this generic review, e.g., reactor trip on upsets in the secondary cooling system of the plant, additional operator training, improvements in auxiliary feedwater reliability, and further analyses of small break loss-of-coolant accidents. Other recommendations for longer term improvements are specified in the report.

The staff believes implementation of the recommendations stated in this report would further increase the safety margins in the B&W pressurized-water-reactor (PWR) plants. Certain of these recommendations also apply to the other PWR vendors (Westinghouse and Combustion Engineering) as well as to boiling-water-reactor (BWR) plants designed by the General Electric Company.

TVA NUCLEAR PROGRAM REVIEW

On Mar. 29, 1979, at the direction of the TVA Board of Directors, the TVA General Manager directed the TVA staff to follow closely the NRC investigation of the Three Mile Island nuclear plant accident, to review equipment, design, procedures, and staffing in TVA's nuclear plants, and to report its findings to the Board as soon as possible. Accordingly a task force was appointed and conducted an intensive review of the TVA nuclear program to determine the lessons to be learned from the Three Mile Island accident, the current status of the TVA safety program in light of those lessons, and changes that could be made to improve the program.

The task force completed its study and issued a draft report[17] in May 1979. The recommendations in this report, which were subsequently adopted by the TVA Board, encompassed all aspects of TVA's nuclear power program, e.g., (1) organization and management of the nuclear power program, (2) operator selection, training, and staffing, (3) radiation standards for employees, (4) nuclear plant design and operation, and (5) emergency planning.

Several improvements were recommended in organization and management of the program. The most significant is the formation of an independent safety review staff, which is outside the power, construction, and design organizations and which has direct access to the TVA Board of Directors. Another very important recommendation is to create a separate organization for nuclear power generation within the Office of Power to concentrate on its unique problems, and to consolidate nuclear safety and nuclear engineering functions into a new nuclear engineering branch in the design organization.

Significant changes in the selection and training program for nuclear power-plant operators are recommended. These changes will require more stringent intelligence tests in the recruiting of candidates, more intensive training over several years, and an increase in salary to a level that will recognize the professional status of the nuclear plant operator. Current operators and trainees will be required to meet the new standards.

Nuclear plant support will be improved by the recommendation that an emergency response team be dispatched to a plant experiencing an emergency to assist in managing operations and in communicating with other TVA experts.

The exposure of TVA employees to radiation in nuclear plants will be reduced with the recommendation that the current TVA limit of 5 rems/year be lowered to 4 rems/year.

Emergency planning will be improved with the recommendation that contingency planning for evacuating or sheltering people in the event of a nuclear accident be extended out to 16 km.

Several recommended design changes will improve the safety of TVA nuclear plants. The changes include the following:

1. Improvement in status monitoring of main-line process components.
2. Capability of venting the primary system.
3. Installation of redundant reactor vessel water-level indicators.
4. Use of an advanced (computer-processed and video-displayed) core monitoring system.
5. Modifications in containment isolation both to ensure isolation in a general emergency and to close individual waste water lines when high radiation is detected.
6. Improvements in sampling and radiation monitoring systems to enhance sampling capability following an accident.
7. Modifications (where not already effected) to provide better relief valves and valve-position indication.

Schedules have been established for the implementation of these design changes on the various TVA nuclear plants—both plants in operation and those still under construction. The task force report also recommended additional design effort in small-break analysis and increased operational feedback, using the NRC Licensee Event Reports.

In addition to the preceding recommendations, the task force also noted some 16 other areas where TVA had previously initiated safety-related improvements, some 5 of which might have been involved in an accident of the Three Mile Island type if the improvements had not been made.

RADIOACTIVITY SAMPLING[18]

Emergency actions by the Food and Drug Administration (FDA), as part of the interagency response to the Three Mile Island accident, took place throughout April and will continue as long as needed. The environmental monitoring effort by the FDA included extensive field work by the Office of Regional Operations (EDRO) in conjunction with planning efforts of the Bureau of Radiological Health. EDRO personnel devoted a total of 787 man-days to the collection of about 1300 milk, food, and water samples between March 29 and April 30.

EDRO arranged daily for shipment of samples, by air, to FDA's Winchester Engineering and Analytical Center (WEAC) in Massachusetts. WEAC analysis of the 432 food samples and 113 water samples revealed no measurable ^{131}I. Of the 760 milk samples analyzed by WEAC, 49 were found to contain measurable amounts of ^{131}I. However, the maximum level for ^{131}I was 36 pCi/liter, a level which is at least 300 times below the guides recommended by FDA for protective action. Thirty-four milk samples also contained trace amounts of ^{137}Cs, but this is believed to be fallout from earlier nuclear atmospheric testing rather than releases from the reactor. Analysis of each sample requires nearly 1 h; consequently WEAC personnel operated around the clock.

The results of all sample analyses were entered in the computer at the Parklawn Building in Rockville, Md., using a remote terminal at WEAC and Bureau-designed software. The EDRO office in Philadelphia maintained a constant link to this data bank through its own remote terminal, as did the Commonwealth of Pennsylvania office in Harrisburg. The Environmental Protection Agency (EPA) supplied data from its independent sampling of milk and water in Pennsylvania, both before and after the accident, which were stored in a separate file in the Parklawn computer.

With respect to the overall federal monitoring program in response to the accident, a memorandum from the White House assigned lead responsibility to the EPA on April 13. EPA was charged with "responsibility for coordinating the collection and documentation of the environmental radiation data collected by all of the Federal agencies involved since the accident occurred on March 28, 1979." EPA was also directed to continue to maintain an operations center in the vicinity of Three Mile Island and to continue monitoring airborne and waterborne radioactivity.

In the same memorandum, FDA was directed to "continue to conduct radioanalyses of milk and food in the vicinity of Three Mile Island at appropriate intervals." The Bureau of Radiological Health is continuing to maintain two staff members on Three Mile Island as part of the team of FDA, EPA, and NRC personnel participating in environmental sampling and decontamination.

LIABILITY INSURANCE PAYMENTS[19]

Representatives of American Nuclear Insurers (ANI) were dispatched to the area near the Three Mile

Island nuclear power plant in Pennsylvania immediately after the accident occurred on March 28.

ANI and the Mutual Atomic Energy Liability Underwriters (MAELU) opened an insurance service office on March 30 to provide reimbursement for living expenses of families that evacuated the area on the recommendation of Pennsylvania Governor Richard Thornburgh. The Governor had urged that families with pregnant women or preschool age children leave the area within a 5-mile radius of the plant.

As of June 1, more than 3000 families had received $1.2 million in payments from ANI and MAELU as reimbursement for emergency living expenses incurred by their evacuation. In addition to these payments, more than $40,000 has been provided to those people who lost wages as a result of their evacuation from the 5-mile area.

At the peak of the operation, 51 claims representatives recruited from member companies of the pools were on the scene processing applications. A staff of claims representatives remains in Harrisburg, and a claims coordinator has joined ANI in Farmington, Conn., to supervise the handling of claims.

ANI staff engineers estimate the total property loss of the Three Mile Island plant to be $140 million. An ANI engineer continues to remain at the plant site to observe cleanup operations.

MISCELLANEOUS ACTIONS OF NOTE

1. **Mitchell Rogovin To Direct NRC's Three Mile Island Inquiry.**[20] Mitchell Rogovin, a senior partner in the Washington law firm of Rogovin, Stern, and Hugh, has been appointed by the NRC to direct a special independent inquiry into the March 28 accident at the Three Mile Island nuclear power plant in Pennsylvania. Rogovin and his law firm will direct the inquiry to determine actual events that occurred at Three Mile Island and their causes and the actions of the licensee, Metropolitan Edison Company, and NRC personnel before and after the accident. The inquiry will also identify areas of deficiency revealed by the accident and areas in which further investigation is warranted. It is expected that the special inquiry will take about 6 months.

2. **HEW To Conduct Studies of Potential Health Effects.**[21] Four studies to be conducted by the Department of Health, Education, and Welfare, in cooperation with the Pennsylvania State Health Department, include (1) record names, vital statistics and relevant health data for the 50,000 people living within 5 miles of Three Mile Island, (2) study every pregnant woman and their offsprings within 10 miles, (3) determine social and psychological impact, and (4) establish a registry of plant workers and their exposures.

3. **NRC To Prepare Environmental Impact Statement on Three Mile Island Waste Water.**[22] The NRC staff has been directed to prepare an environmental assessment regarding proposals to decontaminate and dispose of radioactive contaminated waste water from the Three Mile Island facility.

4. **Chief Counsel of President's Three Mile Island Commission Resigned.**[23] Citing personal reasons, Ronald B. Natalie, Chief Counsel of the President's Commission on Three Mile Island, resigned on June 15. He was replaced by his deputy, Stanley M. Gorinson, who came to the NRC from the Department of Justice.

5. **NRC Issues Three Mile Island Bibliography.**[24] The NRC has published a Title List of Publicly Available Documents, Three Mile Island Unit 2, Docket 50-320, as NRC Report NUREG-0568. The documents listed are cumulative to May 21, 1979, and the report is available from NTIS.

REFERENCES

1. W. R. Casto and Wm. B. Cottrell, Preliminary Report on the Three Mile Island Incident, *Nucl. Safety,* 20(4): 483-490 (July–August 1979).
2. *At. Energy Clearing House,* 25(22): 74 (May 28, 1979).
3. Letter to Joseph M. Hendrie, NRC Chairman, from Max W. Carbon, ACRS Chairman, Interim Report No. 2 on Three Mile Island Nuclear Station Unit 2, May 16, 1979.
4. Letter to Joseph M. Hendrie, NRC Chairman, from Max W. Carbon, ACRS Chairman, Interim Report No. 3 on Three Mile Island Nuclear Station Unit 2, May 16, 1979.
5. *At. Energy Clearing House,* 29(19): 32-46 (May 7, 1979).
6. *At. Energy Clearing House,* 25(21): 47-53 (May 21, 1979).
7. *At. Energy Clearing House,* 25(22): 34-56 (May 28, 1979).
8. *At. Energy Clearing House,* 25(23): 32-46 (June 4, 1979).
9. *At. Energy Clearing House,* 25(24): 2-21 (June 11, 1979).
10. *At. Energy Clearing House,* 25(24): 22-28 (June 11, 1979).
11. *At. Energy Clearing House,* 25(20): 4-15 (May 14, 1979).
12. *At. Energy Clearing House,* 25(21): 2-9 (May 21, 1979).
13. *At. Energy Clearing House,* 25(21): 10-46 (May 21, 1979).
14. *At. Energy Clearing House,* 25(22): 57-73 (May 28, 1979).
15. Letter to Boyce Greir, NRC Director, Region 1, from J. G. Herbein, Vice President, Metropolitan Edison Company, Three Mile Island Nuclear Station Unit 2 (TMI-2) License No. DPR-73, Docket No. 50-320, May 15, 1979.
16. *Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock & Wilcox Company,* NRC Report NUREG-0560, NTIS, May 1979.

17. D. G. Powell et al., *TVA Nuclear Program Review*, unnumbered draft report, Tennessee Valley Authority, May 1979.
18. *Bureau of Radiological Health Bulletin*, Vol. XIII, No. 9, May 14, 1979.
19. *American Nuclear Insurers Letter*, Vol. 3, No. 6, June 1979.
20. NRC Press Release No. 79-104, June 14, 1979.
21. *At. Energy Clearing House*, 25(23): 31 (June 4, 1979).
22. *At. Energy Clearing House*, 25(23): 56 (June 4, 1979).
23. *At. Energy Clearing House*, 25(25): 20 (June 18, 1979).
24. *Power Reactor Events*, Vol. 1, No. 2, Nuclear Regulatory Commission, May 1979.

Summary of TMI-2 Lessons Learned Task Force Report

Adapted by the *Nuclear Safety* Staff

[Editor's Note: The ramifications of the Mar. 28, 1979, accident at the Three Mile Island 2 (TMI-2) nuclear power plant are expected to be felt in the nuclear industry for a long time. The two preceding issues of *Nuclear Safety* published information then available on the accident, including the preliminary report,[1] subsequent developments,[2] and the public dose.[3] In the first several months after the accident, numerous investigations were initiated, several of which have already (as of Sept. 1, 1979) been concluded.[4-6] *Nuclear Safety* will carry a summary report of the accident encompassing these reports after the release of the report by the President's Commission on the Accident at Three Mile Island (expected before the end of this calendar year). In the meantime, we have excerpted here the findings[4] of the TMI-2 Lessons Learned Task Force, which was established in May by Harold R. Denton, Director of the Nuclear Regulatory Commission (NRC) Office of Nuclear Reactor Regulation. This task force was charged with the job of reviewing the accident and recommending possible regulatory improvements, which might then be implemented by changes in legislation, regulations, staff positions, review procedures, or otherwise.]

Abstract: *After its review of the Three Mile Island 2 accident, the TMI-2 Lessons Learned Task Force recommended that a number of actions in the areas of design and analysis and plant operations be required in the short term to provide substantial additional protection for the public health and safety. All nuclear power plants in operation or in various stages of construction or licensing action are affected to varying degrees by the specific recommendations. Comments by the Advisory Committee on Reactor Safeguards concerning the short-term recommendations are presented.*

The TMI-2 Lessons Learned Task Force is one of several activities related to the Three Mile Island accident now under way in the Nuclear Regulatory Commission (NRC) Office of Nuclear Reactor Regulation (NRR). The purpose of the task force is to identify and evaluate those safety concerns originating with the TMI-2 accident which require licensing actions [beyond those already specified in IE (Inspection and Enforcement) Bulletins and NRC orders] for presently operating reactors as well as for pending operating-license and construction-permit applications. In making this evaluation, the task force will review and evaluate investigative information; staff evaluations of responses to IE Bulletins and NRC orders; recommendations of the Commissioners, the Advisory Committee on Reactor Safeguards (ACRS), and the NRC staff; the recommendations in Report NUREG-0560 *(Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock & Wilcox Company)*; and those from outside the NRC. In addition, the task force is charged with the job of identifying, analyzing, and recommending changes in the licensing requirements and the licensing process for nuclear power plants based on the lessons learned.

Actions recommended by the Lessons Learned Task Force and approved by the NRR Director or the NRC, as appropriate, will be assigned to the Division of Project Management, Division of Systems Safety, Division of Operating Reactors, and the Bulletins and Orders (B&O) Task Force for implementation on pending license applications and on operating plants. At that time the appropriate licensing boards will be formally notified of these licensing matters.

The short-term actions recommended by the Lessons Learned Task Force in this report, when combined with the requirements associated with implementation of the IE Bulletins on TMI-2, including the generic status reports to be issued by the B&O Task Force, are intended to constitute a sufficient set of short-term requirements to ensure the safety of plants already licensed to operate and those to be licensed for operation in the near future. In addition, commitments from construction-permit applicants to meet these requirements are sufficient bases for the staff to recommend the granting of construction permits for those applications now pending before the hearing boards.

The Lessons Learned Task Force has recommended changes for the short term in light-water nuclear power plants in 12 broad areas (9 in the area of design and analysis and 3 in the operations area). The changes recommended are described below. After approval by the NRR Director or by the NRC, as appropriate, these short-term requirements will be transmitted as licensing requirements to licensees and construction-permit and operating-license applicants. Except as discussed below, the recommended requirements are consistent with existing NRC regulations. Three requirements have been identified, however, which require the revision of present regulations:

1. Inerting all boiling-water-reactor (BWR) containments.

2. Capability to install a recombiner at each light-water-reactor (LWR) facility.

3. Revision of limiting conditions for operation based on safety-system availability.

The first two of the above requirements are governed generally by 10 CFR 50.44 and the last by 10 CFR 50.36.

The Lessons Learned Task Force is therefore recommending that, after approval by the NRR Director, rule-making proceedings be initiated on an immediately effective basis. This method of rule making will permit the prompt imposition of these requirements and will, in items 1 and 2 above, cause existing facilities to comply with the requirements sooner than they would if a proposed rule were published with or without an advance notice of proposed rule making. In item 3 the proposed method will provide a new type of information on operating experience at an earlier time.

DESIGN AND ANALYSIS RECOMMENDATIONS

1. Provide redundant emergency power for the minimum number of pressurizer heaters required to maintain natural-circulation conditions in the event of loss of off-site power. Also, provide emergency power to the control and motive power systems for the power-operated relief valves and associated block valves and to the pressurizer-level indication instrument channels.

2. Provide performance verification by full-scale prototypical testing for all relief and safety valves. Test conditions will include two-phase slug flow and subcooled liquid flow calculated to occur for design-basis transients and accidents.

3a. Provide in the control room either a reliable, direct position indication for the valves or reliable flow-indication devices downstream of the valves.

3b. Perform analyses and implement procedures and training for prompt recognition of low reactor coolant level and inadequate core cooling using existing reactor instrumentation (flow, temperature, power, etc.) or short-term modifications of existing instruments. Describe further measures and provide supporting analyses that will yield more direct indication of low reactor coolant level and inadequate core cooling, such as reactor vessel water-level instrumentation.

4. Provide containment isolation on diverse signals in conformance with Section 6.2.4 of the Standard Review Plan, review isolation provisions for nonessential systems and revise as necessary, and modify containment isolation designs as necessary to eliminate the potential for inadvertent reopening after the isolation signal has been reset.

5a. For plants that have external recombiners or purge systems, provide dedicated penetrations and isolation systems that meet the redundancy and single-failure requirements of the NRC regulations. Modify the design as necessary so that these systems are not connected to, or are branch lines of, the large containment purge penetrations.

5b. Provide inerting for all Mark I and Mark II BWR containments. This would require changes at Vermont Yankee and Hatch 2 (operating plants) and in pending operating-license applications for Mark I and II BWRs.

5c. A minority of the task force recommends that all operating reactors that do not already have the capability be required to provide the capability to add, within a few days after an accident, a hydrogen recombiner system for postaccident hydrogen control.

6a. Perform leakage-rate tests on systems outside the containment which process primary coolant and could contain high-level radioactive materials. Develop and implement a periodic testing program and preventive maintenance programs.

6b. Perform a design review of the shielding of systems processing primary coolant outside the containment. Determine any areas or equipment that are vital for postaccident occupancy or operation and assure that access and performance will not be unduly impaired due to radiation from these systems.

7a. Provide automatic initiation of all auxiliary feedwater systems. The initiation signals and circuits shall be designed in such a manner that a single failure will not result in the loss of auxiliary feedwater system function. Testability of the initiating signals and circuits shall be a feature of the design. The initiating signals and circuits shall be powered from the emergency buses. Manual capability to initiate the auxiliary feedwater system from the control room must be retained and must be implemented in such a manner that a single failure in the manual circuits will not result in the loss of system function. The alternating-current motor-driven pumps and valves in the auxiliary feedwater system must be included in the automatic actuation (simultaneous or sequential) of the loads to the emergency buses. The design of the automatic initiating signals and circuits must be such that their failure will not result in the loss of manual capability to initiate the auxiliary feedwater system from the control room.

7b. Provide safety-grade indication in the control room of auxiliary feedwater flow for each steam generator. The flow instrument channels shall be powered from the emergency buses, consistent with satisfying the power diversity requirements for auxiliary feedwater systems.

8a. Review and upgrade the capability to obtain samples from the reactor coolant system and containment atmosphere under high-radioactivity conditions. Provide the capability for chemical and spectrum analysis of high-level samples on site.

8b. Provide high-range radiation monitors for noble gases in plant effluent lines and a high-range radiation monitor in the containment. For the monitoring of effluent release lines, provide instrumentation capable of measuring and identifying radioiodine and particulate radioactive effluents under accident conditions.

8c. Provide instrumentation for accurately determining in-plant airborne radioiodine concentrations to minimize the need for unnecessary use of respiratory protection equipment.

9a. Provide the analysis, emergency procedures, and training to substantially improve operator performance during a small-break loss-of-coolant accident.

9b. Provide the analysis, emergency procedures, and training needed to assure that the reactor operator can recognize and respond to conditions of inadequate core cooling.

9c. Provide the analysis, emergency procedures, and training to substantially improve operator performance during transients and accidents, including events that are caused or worsened by inappropriate operator actions.

OPERATIONS RECOMMENDATIONS

1a. Review plant administrative and management procedures. Revise procedures as necessary to assure that reactor operations command and control responsibilities and authority are properly defined. Corporate management shall revise and promptly issue an operations policy directive that emphasizes the duties, responsibilities, and authority and lines of command of the control-room operators, the shift technical advisor, and the person responsible for reactor operations command in the control room (i.e., the senior reactor operator).

1b. Provide on each shift at each nuclear power plant a qualified person (the shift technical advisor) with a bachelor's degree or the equivalent in a science or engineering discipline and with specific training in the plant response to off-normal events and in accident analysis of the plant. Shift technical advisors shall serve in an advisory capacity to shift supervisors. The licensee shall assign normal duties to the shift technical advisor which pertain to the engineering aspects of assuring safe operation of the plant, including the review and evaluation of operating experience.

1c. Review and revise plant procedures as necessary to assure that a shift turnover checklist is provided and required to be completed and signed by the on-coming and off-going individuals responsible for the command of operations in the control room. Supplementary checklists and shift logs should be developed for the entire operations organization, including instrument technicians, auxiliary operators, and maintenance personnel.

2a. Review plant emergency procedures, and revise as necessary, to assure that access to the control room under normal and accident conditions is limited to those persons necessary to the safe command and control of operations.

2b. A separate technical support center shall be provided for use by plant management, technical, and engineering support personnel. In an emergency this center shall be used to assess plant status and potential off-site impact in support of the control-room command and control function. The center should also be used in conjunction with implementation of on-site and off-site emergency plans, including communications with an off-site emergency-response center. Provide at the on-site technical support center the as-built drawings of general plant arrangements and piping, instrumentation, and electrical systems. Photographs of as-built system layouts and locations may be an acceptable method of satisfying some of these needs.

2c. Each operating nuclear power plant should establish and maintain a separate on-site operational support center outside the control room. In the event of an emergency, shift support personnel (e.g., auxiliary operators and technicians) other than those required and allowed in the control room shall report to the center for further orders and assignment.

3. Require that the Technical Specifications for each reactor provide that the reactor be placed in a hot-shutdown condition within 8 h and in a cold-shutdown condition within 24 h of any time that it is found to be or to have been in operation with a complete loss of safety function (e.g., loss of emergency feedwater, high-pressure emergency core-cooling system (ECCS), low-pressure ECCS, containment,

emergency power, or other prescribed safety function). Require that the cause of the loss of safety function be assessed (e.g., maintenance, operations error) and that an evaluation of alternative corrective actions be made and documented by the licensee. Require that the senior corporate officer responsible for operation of the facility present the licensee's recommendation for corrective action and evaluation of the alternatives at a public meeting with senior NRC officials. Require that the senior NRC officials issue their decision at the public meeting, or at a subsequent public meeting if time is required for staff evaluation, concerning the adequacy of the changes to improve operational reliability proposed by the utility. Allow the facility to return to power only after NRC approval of the changes proposed by the licensee.

IMPLEMENTATION OF SHORT-TERM RECOMMENDATIONS

The Lessons Learned Task Force set up a schedule for implementing the recommendations for the various categories of plants. The first category consists of plants that have construction permits and plants in construction-permit review. They are required to make a commitment to conform to the short-term recommendations. The second category consists of operating plants and plants in operating-license review. Provision for implementation of the recommendation of the short-term requirements for these latter plants in two phases is shown in Table 1.

The intent of the two-phase implementation is to provide maximum timely improvements in safety consistent with practical limitations on the ability of licensees and applicants to design, procure, and install equipment or to develop and implement administrative changes. The task force has categorized the short-term recommendations into those which are essentially procedural (category A) and can therefore be implemented expeditiously (prior to Jan. 1, 1980) and those which involve design changes and/or hardware procurement and installation (category B) and will require a longer time period (prior to Jan. 1, 1981). Although a longer time period is allowed for category B items, the task force believes that many of these changes can and should be accomplished within a shorter time frame. To this end, the task force also recommends that meetings be scheduled with all operating-plant licensees and applicants for operating licenses to establish plant-specific schedules for the category B items. At these meetings the need for providing scheduled relief in specific instances for good cause will also be

Table 1 Implementation of Short-Term Recommendations for Operating Plants and Plants in Operating-License Review

Position		Implementation category*
Abbreviated title	Position description	
Emergency power supply requirement	Complete implementation	A
Relief and safety valve testing	Submit program description and schedule	A
	Complete test program	By July 1981†
Direct indication of valve position	Complete implementation	A
Instrumentation for inadequate core cooling	Develop procedures and describe existing instrumentation	A
	New instrumentation design, subcooling meter installation, and implementation schedule	A
	Complete installation of new instrumentation	B
Diverse containment isolation	Complete implementation	A
Dedicated H_2 control penetrations	Description and implementation schedule	A
	Complete installation	B
Rule making to require inerting BWR containments	Inert Vermont Yankee and Hatch 2	
	Design and equipment to inert new Mark I and II containments	‡
	Inert new Mark I and II containments	‡
Combustible gas control recombiner	Rule making to require capability of installing recombiners	‡
	Review procedures and bases for recombiner use	B
Systems integrity for high radioactivity	Immediate leak reduction program	A
	Preventive maintenance program	A
Plant shielding review	Complete the design review	A
	Implement plant modifications	B
Automatic initiation of auxiliary feed	Complete implementation of control grade	A
	Complete implementation of safety grade	B

Table 1 (Continued)

Position		Implementation category*
Abbreviated title	Position description	
Auxiliary feed flow indication	Complete implementation	A
Postaccident sampling	Design review complete	A
	Preparation of revised procedures	A
	Implement plant modifications	B
	Description of proposed modification	A
High-range effluent monitor	Installation complete	B
Improved iodine instrumentation	Complete implementation	A
Transient and accident analysis	Complete analyses, procedures, and operator training	§
Shift supervisor responsibilities	Complete implementation	A
Shift safety engineer	Shift technical advisor on duty	A
	Complete training	B
Shift turnover procedures	Complete implementation	A
Control-room access control	Complete implementation	A
On-site technical support center	Establish center	A
	Upgrade to meet all requirements	B
On-site operational support center	Complete implementation	A
Rule making to revise limiting conditions for operation for safety-system availability	Technical specification change	‡

*Category A: Implementation complete by Jan. 1, 1980, or prior to operating license.

Category B: Implementation complete by Jan. 1, 1981.

†After July 1, 1982, relief- and safety-valve testing will be satisfactorily completed for all plants prior to receiving an operating license.

‡Implementation schedules will be established by the NRC in the course of the immediately effective rule making. The task force recommends that the rule-making process be initiated promptly.

§Analyses, procedural changes, and operator training will be provided by all operating plant licensees and applicants for operating licenses following the schedule in Table 2.

considered. Table 1 identifies the specific category of each of the short-term recommendations.

For all plants in construction-permit review and all plants under construction for which an operating-license application has not yet been presented, the applicant/construction-permit holder shall provide a commitment to comply with the recommendations of this report within 30 days after receiving a letter from the NRC Office of Nuclear Reactor Regulation specifying the particular licensing requirements that apply to each particular plant design. All requirements of each position shall be incorporated into the plant design, as appropriate, and described in the Final Safety Analysis Report when an application for an operating license is presented.

For operating plants, implementation of the recommendations shall be in two phases, as specified in Table 1. Category A items shall be implemented prior to Jan. 1, 1980, and category B items prior to Jan. 1, 1981, with the exception of the safety and relief valve qualification testing (July 1, 1981). For plants with tendered operating-license applications, category A items shall be implemented prior to receipt of an operating license. Specific schedules for the category B items will be developed in meetings with licensees and applicants to be conducted within 30 days of notification.

Table 2 indicates the proposed schedule for analyses, procedural changes, and operator training for all operating plant licensees and applicants for operating licenses.

ACRS REPORT ON SHORT-TERM RECOMMENDATIONS

The NRC has received from its ACRS a report concerning an ACRS review of the short-term recommendations of the NRC's TMI-2 Lessons Learned Task Force. This report is presented in its entirety below.

> During its 232nd meeting, August 9–11, 1979, the Advisory Committee on Reactor Safeguards completed a review of the short-term recommendations of the TMI-2 Lessons Learned Task Force as reported in NUREG-0578. These recommendations had been reviewed, in part, by an ACRS Subcommittee at a meeting in Washington, D. C., on July 27, 1979. During its review the Committee had the benefit of discussions with members of the Task Force. Comments from representatives of the nuclear industry were also considered.
>
> In its review, the Committee has noted that the recommendations in NUREG-0578 are those deemed by the Task Force to be required in the short term to provide substantial additional protection for the public health and safety.
>
> The Committee has considered both the recommendations themselves and the schedules proposed

Table 2 Tasks and Responsibilities Timetable*

Task description	Completion date
1. Small-break LOCA analysis and preparation of emergency procedure guidelines	July–September 1979†
2. Implementation of small-break LOCA emergency procedures and retraining of operators	Dec. 31, 1979
3. Analysis of inadequate core cooling and preparation of emergency procedure guidelines	October 1979
4. Implementation of emergency procedures and retraining related to inadequate core cooling	January 1980
5. Analysis of accidents and transients and preparation of emergency procedure guidelines	Early 1980
6. Implementation of emergency procedures and retraining related to accidents and transients	3 months after guidelines established
7. Analysis of LOFT small-break tests	Pretest (mid-September 1979)

*LOCA = loss-of-coolant accident.
LOFT = Loss-of-Fluid Test.
†Range covers completion dates for the four nuclear steam-system suppliers.

for their implementation. Regarding the latter, the Committee believes that the orderly and effective implementation and the appropriate level of review and approval by the NRC Staff will require a somewhat more flexible, and in some cases more extended, schedule than is implied by NUREG-0578.

With regard to the requirements themselves, the Committee agrees with the intent and substance of all except those discussed below.

2.1.5 Post-Accident Hydrogen Control Systems

a. The Committee agrees with the recommendations relating to dedicated penetrations for external recombiners or purge systems for operating plants that have such systems.

b. and c. The majority of the Task Force has recommended rule-making to require inerting of BWR Mark I and II reactors. A minority of the Task Force has recommended rule-making to require that all operating light water reactors provide the capability to use a hydrogen recombiner.

The Committee believes that questions relating to hydrogen generation during and following an accident, the rate and amount of generation, the need to control it, and the means of doing so, need to be reexamined. The Task Force has advised the Committee that it is considering this question further in connection with its longer-term recommendations which are scheduled to be completed by September 1979. The ACRS believes that decisions concerning possible additional measures to deal with hydrogen should be deferred pending early evaluation of the forthcoming longer-term Task Force recommendations.

2.1.8 Instrumentation to Follow the Course of an Accident

With regard to instrumentation to follow the course of an accident, the ACRS believes that containment pressure, containment water level, and on-line monitoring of hydrogen concentration in the containment should also be considered for implementation for all operating reactors on the same schedule as that recommended by the Lessons Learned Task Force.

2.2.1.b Shift Technical Advisor

The Committee agrees completely with the two closely related objectives of this recommendation. One relates to the presence in the control room during off-normal events of an individual having technical and analytical capability and dedicated to concern for safety of the plant. The other relates to the need for an on-site, and perhaps dedicated, engineering staff to review and evaluate safety-related aspects of plant design and operation. The achievement of these objectives will contribute significantly to the safe operation of a plant.

The Committee believes that there may be difficulty in finding a sufficient number of people with the required qualifications and interest in shift work to fill the Technical Advisor positions. The Committee therefore believes the solution proposed by the Staff should not be mandatory but that alternate solutions also should be considered.

2.2.3 Revised Limiting Conditions for Operation

The Committee agrees with the findings of the Task Force that there are too many human or operational errors resulting in the defeat of an entire safety system, that the number of such occurrences should be and can be reduced, and that the ultimate responsibility for doing this must rest with the licensee.

The Committee, however, is not convinced that the Task Force proposal is the best or only way to increase the licensee's awareness of the need to improve operational reliability, and suggests that measures short of shutdown, such as a rule that requires actions similar to those of a show-cause order, may be equally effective.

REFERENCES

1. W. R. Casto and Wm. B. Cottrell, Preliminary Report on the Three Mile Island Incident, *Nucl. Safety*, 20(4): 483-492 (July–August 1979).
2. Ad Hoc Population Dose Assessment Group, Preliminary Dose and Health Impact of the Accident at the Three Mile Island Nuclear Station, *Nucl. Safety*, 20(5): 591-594 (September–October 1979).
3. Wm. B. Cottrell, Developments Pertaining to the Three Mile Island Accident, *Nucl. Safety*, 20(5): 613-623 (September–October 1979).
4. Nuclear Regulatory Commission, *TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations*, NRC Report NUREG-0578, NTIS, July 1979.
5. Nuclear Regulatory Commission, Office of Inspection and Enforcement, *Investigation into the March 28, 1979, Three Mile Island Accident by the Office of Inspection and Enforcement*, NRC Report NUREG-0600 (Investigative Report 50-320/79-10), NTIS, August 1979.
6. Nuclear Safety Analysis Center (operated by Electric Power Research Institute), *Analysis of Three Mile Island Unit 2 Accident*, Report NSAC-1, Electric Power Research Institute, Palo Alto, Calif., July 1979.
7. Letter to J. M. Hendrie, NRC Chairman, from M. W. Carbon, ACRS Chairman, *Short-Term Recommendations of the TMI-2 Lessons Learned Task Force*, Aug. 13, 1979.

Report of the President's Commission on the Accident at Three Mile Island

Adapted by the *Nuclear Safety* Staff

[Editor's Note: The *Report of the President's Commission on the Accident at Three Mile Island,* which was released in the latter part of October 1979, is subtitled *The Need for Change: The Legacy of TMI.* The table of contents of the report is as follows: Preface, Overview, Commission Findings, Commission Recommendations, Account of the Accident, Appendices: Executive Order, Commission Operations and Methodology, Commissioners' Biographies, Staff List, and Glossary. The Preface and the Commission Recommendations are reprinted here in their entirety.]

PREFACE

The Charge to the Commission

On March 28, 1979, the United States experienced the worst accident in the history of commercial nuclear power generation. Two weeks later the President of the United States established a Presidential Commission. The President charged the 12-member Commission as follows:

The purpose of the Commission is to conduct a comprehensive study and investigation of the recent accident involving the nuclear power facility on Three Mile Island in Pennsylvania. The Commission's study and investigation shall include:

(a) a technical assessment of the events and their causes; this assessment shall include, but shall not be limited to, an evaluation of the actual and potential impact of the events on the public health and safety and on the health and safety of workers;

(b) an analysis of the role of the managing utility;

(c) an assessment of the emergency preparedness and response of the Nuclear Regulatory Commission and other federal, state, and local authorities;

(d) an evaluation of the Nuclear Regulatory Commission's licensing, inspection, operation, and enforcement procedures as applied to this facility;

(e) an assessment of how the public's right to information concerning the events at TMI was served and of the steps which should be taken during similar emergencies to provide the public with accurate, comprehensible, and timely information; and

(f) appropriate recommendations based upon the Commission's findings.

The Accident

At 4:00 a.m. on March 28, 1979, a serious accident occurred at the Three Mile Island Unit 2 (TMI-2) nuclear power plant near Middletown, Pa. The accident was initiated by mechanical malfunctions in the plant and made much worse by a combination of human errors in responding to it. During the next 4 days the extent and gravity of the accident were unclear to the managers of the plant, to federal and state officials, and to the general public. What is quite clear is that its impact, nationally and internationally, has raised serious concerns about the safety of nuclear power. This Commission was established in response to those concerns.

What We Did

The investigation of the Commission was made by our able and hard-working staff. We also had the help of a number of consultants and commissioned several studies. It is primarily due to the work of the staff that we accomplished the following.

To determine what happened and why, we examined with great care the sequence of events that occurred during the accident. We have attempted to evaluate the significance of various equipment failures as well as the importance of actions (or failures of actions) on the part of individuals and organizations.

We analyzed the various radiation releases and came up with the best possible estimates of the health effects of the accident. In addition, we checked into how well the health and safety of the workers were protected during normal operating conditions and how well their health and safety and that of the general public would have been protected in the case of a more serious accident.

We conducted an in-depth examination of the role played by the utility and its principal suppliers. We examined possible problems of organization, procedures, and practices which might have contributed to the accident. Since the major cause of the accident was due to inappropriate actions by those who were operating the plant and supervising that operation, we looked very carefully at the training programs that prepare operators and the procedures under which they operate.

As requested by the President, we examined the emergency plans that were in place at the time of the accident. We also probed the responses to the accident by the utility, by state and local governmental agencies in Pennsylvania, and by a variety of federal agencies. We looked for deficiencies in the plans and in their execution so we can make recommendations for improvements for any future accident. In this process we had in mind how well the response would have worked if the danger to public health had been significantly greater.

We examined the coverage of the accident by the news media. This was a complex process in which we had to determine whether errors in media accounts were due to ignorance or confusion on the part of the official sources, to the way they communicated this information to the media, or to mistakes committed by the reporters themselves. We examined what sources were most influential on the people who needed immediate information and how well the public was served by the abundant coverage that was provided. We also attempted to evaluate whether or not the coverage tended to exaggerate the seriousness of the accident either by selectively using alarming quotes more than reassuring ones, or through purposeful sensationalism.

Finally, we spent a great deal of time on the agency that had a major role in all the preceding: the Nuclear Regulatory Commission (NRC). The President gave us a very broad charge concerning this agency. We therefore tried to understand its complex structure and how well it functions, its role in licensing and rule making, how well it carries out its mission through its inspection and enforcement program, the role it plays in monitoring the training of operators, and its participation in the response to the emergency, including the part it played in providing information to the public.

We took more than 150 formal depositions and interviewed a significantly larger number of individuals. At our public hearings we heard testimony under oath from a wide variety of witnesses. We collected voluminous material that will fill about 300 ft of shelf space in a library. All this material will be placed into the National Archives. The most important information extracted from this in each of the areas will appear in a series of "Staff Reports to the Commission."

On the basis of all this information, the Commission arrived at a number of major findings and conclusions. In turn, these findings led the Commission to a series of recommendations responsive to the President's charge.

At the beginning of this volume will be found an overview of our investigation, followed by those findings and recommendations which commanded a significant consensus among the members of the Commission. Each recommendation was approved by a majority of Commissioners.

What We Did Not Do

It is just as important for the reader to understand what the Commission did *not* do.

Our investigation centered on one accident at one nuclear power plant in the United States. Although acting under the President's charge, we had to look at a large number of issues affecting many different organizations; there are vast related issues which were outside our charge and which we could not possibly have examined in a 6-month investigation.

We did not examine the entire nuclear industry. (Although, through our investigation of the NRC, we have at least some idea of the standards being applied to it across the board.) We have not looked at the military applications of nuclear energy. We did not consider nuclear weapons proliferation. We have not dealt with the question of the disposal of radioactive waste or the dangers of the accumulation of waste fuel within nuclear power plants adjacent to the containment buildings. We made no attempt to examine the entire fuel cycle, starting with the mining of uranium. And, of course, we made no examination of the many other sources of radiation, both natural and man-made, that affect all of us.

We have not attempted to evaluate the relative risks involved in alternate sources of energy. We are aware of a number of studies that try to do this. We are also aware that some of these studies are subjects of continuing controversy.

We did not attempt to reach a conclusion as to whether, as a matter of public policy, the development of commercial nuclear power should be continued or should not be continued. That would require a much broader investigation involving economic, environmental, and political considerations. We are aware that there are 72 operating reactors in the United States with a capacity of 52,000 MW(e). An additional 92 plants have received construction permits and are in various stages of construction. If these are completed, they will roughly triple the present nuclear capacity to generate electricity. This would be a significant fraction of the total U. S. electrical generating capacity of some 600,000 MW. In addition, there are about 200 nuclear power plants in other countries throughout the world.

Therefore the improvement of the safety of existing and planned nuclear power plants is a crucial issue. It is this issue that our report addresses, those changes that can and must be made as a result of the accident—the legacy of Three Mile Island.

COMMISSION RECOMMENDATIONS

A. The Nuclear Regulatory Commission

The Commission found a number of inadequacies in the NRC and, therefore, proposes a restructuring of the agency. Because there is insufficient direction in the present statute, the President and Congress should consider incorporating many of the following measures in statutory form.

Agency Organization and Management

The Commission believes that, as presently constituted, the NRC does not possess the organizational and management capabilities necessary for the effective pursuit of safety goals. The Commission recommends:

1. The NRC should be restructured as a new independent agency in the executive branch.
   a. The present five-member commission should be abolished.
   b. The new agency should be headed by a single administrator appointed by the President, subject to the advice and consent of the Senate, to serve a substantial term (not coterminous with that of the President) in order to provide an expectation of continuity, but at the pleasure of the President to allow removal when the President deems it necessary. The administrator should be a person from outside the present agency.
   c. The administrator should have substantial discretionary authority over the internal organization and management of the new agency and over personnel transfers from the existing NRC. Unlike the present NRC arrangement, the administrator and major staff components should be located in the same building or group of buildings.
   d. A major role of the administrator should be assuring that offices within the agency communicate sufficiently so that research, operating experience, and inspection and enforcement affect the overall performance of the agency.
2. An oversight committee on nuclear reactor safety should be established. Its purpose would be to examine, on a continuing basis, the performance of the agency and of the nuclear industry in addressing and resolving important public safety issues associated with the construction and operation of nuclear power plants, and in exploring the overall risks of nuclear power.
   a. The members of the committee, not to exceed 15 in number, should be appointed by the President and should include: persons conversant with public health, environmental protection, emergency planning, energy technology and policy, nuclear power generation, and nuclear safety; one or more state governors; and members of the general public.
   b. The committee, assisted by its own staff, should report to the President and to Congress at least annually.
3. The Advisory Committee on Reactor Safeguards (ACRS) should be retained, in a strengthened role, to continue providing an independent technical check on safety matters. The members of the committee should continue to be part-time appointees; the Commission believes that the independence and high quality of the members might be compromised by making them full-time federal employees. The Commission recommends the following changes:
   a. The staff of ACRS should be strengthened to provide increased capacity for independent analysis. Special consideration should be given to improving ACRS' capabilities in the field of public health.
   b. The ACRS should not be required to review each license application. When ACRS chooses to review a license application, it should have the statutory right to intervene in hearings as a party. In particular, ACRS should be authorized to raise any safety issue in licensing proceedings, to give reasons and arguments for its views, and to require formal response by the agency to any submission it makes. Any member of ACRS should be authorized to appear and testify in hearings but should be

exempt from subpoena in any proceedings in which he has not previously appeared voluntarily or made an individual written submission.

c. ACRS should have similar rights in rule-making proceedings. In particular, it should have the power to initiate a rule-making proceeding before the agency to resolve any generic safety issue it identifies.

The Agency's Substantive Mandate

The new agency's primary statutory mission and first operating priority must be the assurance of safety in the generation of nuclear power, including safeguards of nuclear materials from theft, diversion, or loss. Therefore the Commission recommends the following:

4. Included in the agency's general substantive charge should be the requirement to establish and explain safety–cost trade-offs; where additional safety improvements are not clearly outweighed by cost considerations, there should be a presumption in favor of the safety change. Transfers of statutory jurisdiction from the NRC should be preceded by a review to identify and remove any unnecessary responsibilities that are not germane to safety. There should also be emphasis on the relationship of the new agency's safety activities to related activities of other agencies. (See recommendations E.2 and F.1.b.)
   a. The agency should be directed to upgrade its operator and supervisor licensing functions. These should include the accreditation of training institutions from which candidates for a license must graduate. Such institutions should be required to employ qualified instructors, to perform emergency and simulator training, and to include instruction in basic principles of reactor science, reactor safety, and the hazards of radiation. The agency should also set criteria for operator qualifications and background investigations, and strictly test license candidates for the particular power plant they will operate. The agency should periodically review and reaccredit all training programs and relicense individuals on the basis of current information on experience in reactor operations. (See recommendations C.1 and C.2.)
   b. The agency should be directed to use a broader definition of matters relating to safety that considers thoroughly the full range of safety matters, including, but not limited to, those now identified as "safety-related" items, which currently receive special attention.
   c. Other safety emphases should include:
      (i) A systems engineering examination of overall plant design and performance, including interaction among major systems and increased attention to the possibility of multiple failures.
      (ii) Review and approval of control room design; the agency should consider the need for additional instrumentation and for changes in overall design to aid understanding of plant status, particularly for response to emergencies (see recommendation D.1).
      (iii) An increased safety research capacity with a broadly defined scope that includes issues relevant to public health. It is particularly necessary to coordinate research with the regulatory process in an effort to assure the maximum application of scientific knowledge in the nuclear power industry.
5. Responsibility and accountability for safe power plant operations, including the management of a plant during an accident, should be placed on the licensee in all circumstances. It is, therefore, necessary to assure that licensees are competent to discharge this responsibility. To assure this competency and in light of our findings regarding Metropolitan Edison, we recommend that the agency establish and enforce higher organizational and management standards for licensees. Particular attention should be given to such matters as the following: integration of decision making in any organization licensed to construct or operate a plant; kinds of expertise that must be within the organization; financial capability; quality assurance programs; operator and supervisor practices and their periodic reevaluation; plant surveillance and maintenance practices; and requirements for the analysis and reporting of unusual events.
6. To provide an added contribution to safety, the agency should be required, to the maximum extent feasible, to locate new power plants in areas remote from concentrations of population. Siting determinations should be based on tech-

nical assessments of various classes of accidents that can take place, including those involving releases of low doses of radiation. (See recommendation F.2.)

7. The agency should be directed to include, as part of its licensing requirements, plans for the mitigation of the consequences of accidents, including the cleanup and recovery of the contaminated plant. The agency should be directed to review existing licenses and to set deadlines for accomplishing any necessary modifications. (See recommendations D.2 and D.4.)
8. Because safety measures to afford better protection for the affected population can be drawn from the high standards for plant safety recommended in this report, the NRC or its successor should, on a case-by-case basis, before issuing a new construction permit or operating license:
   a. Assess the need to introduce new safety improvements recommended in this report and in NRC and industry studies.
   b. Review, considering the recommendations set forth in this report, the competency of the prospective operating licensee to manage the plant and the adequacy of its training program for operating personnel.
   c. Condition licensing upon review and approval of the state and local emergency plans.

Agency Procedures

The Commission believes that the agency must improve on prior performance in resolving generic and specific safety issues. Generic safety issues are considered in rule-making proceedings that formulate new standards for categories of plants. Specific safety issues are considered in adjudicative proceedings that determine whether a particular plant should receive a license. Both kinds of safety issues are then dealt with in inspection and enforcement processes. The Commission believes that all these agency functions need improvement and accordingly recommends the following measures:

9. The agency's authorization to make general rules affecting safety should:
   a. Require the development of a public agenda according to which rules will be formulated.
   b. Require the agency to set deadlines for resolving generic safety issues.
   c. Require a periodic and systematic reevaluation of the agency's existing rules.
   d. Define rule-making procedures designed to create a process that provides a meaningful opportunity for participation by interested persons, which ensures careful consideration and explanation of rules adopted by the agency, and which includes appropriate provision for the application of new rules to existing plants. In particular, the agency should: accompany newly proposed rules with an analysis of the issues they raise and provide an indication of the technical materials that are relevant; provide a sufficient opportunity for interested persons to evaluate and rebut materials relied on by the agency or submitted by others; explain its final rules fully, including responses to principal comments by the public, the ACRS, and other agencies on proposed rules; impose, when necessary, special interim safeguards for operating plants affected by generic safety rule making; and conduct systematic reviews of operating plants to assess the need for retroactive application of new safety requirements.
10. Licensing procedures should foster early and meaningful resolution of safety issues before major financial commitments in construction can occur. To ensure that safety receives primary emphasis in licensing, and to eliminate repetitive consideration of some issues in that process, the Commission recommends the following:
    a. Duplicative consideration of issues in several stages of one plant's licensing should, wherever possible, be reduced by allocating particular issues (such as the need for power) to a single stage of the proceedings.
    b. Issues that recur in many licensings should be resolved by rule making.
    c. The agency should be authorized to conduct a combined construction permit and operating license hearing whenever plans can be made sufficiently complete at the construction permit stage.
    d. There should be provision for the initial adjudication of license applications and for appeal to a board whose decisions would not be subject to further appeal to the administrator. Both initial adjudicators and appeal boards should have a clear mandate to pursue any safety issue, whether or not it is raised by a party.
    e. An Office of Hearing Counsel should be established in the agency. This office would

not engage in the informal negotiations between other staff and applicants that typically precede formal hearings on construction permits. Instead, it would participate in the formal hearings as an objective party, seeking to assure that vital safety issues are addressed and resolved. The office should report directly to the administrator and should be empowered to appeal any adverse licensing board determination to the appeal board.

f. Any specific safety issue left open in licensing proceedings should be resolved by a deadline.

11. The agency's inspection and enforcement functions must receive increased emphasis and improved management, including the following elements:

a. There should be an improved program for the systematic safety evaluation of currently operating plants, to assess compliance with current requirements, to assess the need to make new requirements retroactive to older plants, and to identify new safety issues.

b. There should be a program for the systematic assessment of experience in operating reactors, with special emphasis on discovering patterns in abnormal occurrences. An overall quality assurance measurement and reporting system based on this systematic assessment shall be developed to provide: (1) a measure of the overall improvement or decline in safety, and (2) a base for specific programs aimed at curing deficiencies and improving safety. Licensees must receive clear instructions on reporting requirements and clear communications summarizing the lessons of experience at other reactors.

c. The agency should be authorized and directed to assess substantial penalties for licensee failure to report new "safety-related" information or for violations of rules defining practices or conditions already known to be unsafe.

d. The agency should be directed to require its enforcement personnel to perform improved inspection and auditing of licensee compliance with regulations and to conduct major and unannounced on-site inspections of particular plants.

e. Each operating licensee should be subject periodically to intensive and open review of its performance according to the requirements of its license and applicable regulations.

f. The agency should be directed to adopt criteria for revocation of licenses, sanctions short of revocation such as probationary status, and kinds of safety violations requiring immediate plant shutdown or other operational safeguards.

B. The Utility and Its Suppliers

1. To the extent that the industrial institutions we have examined are representative of the nuclear industry, the nuclear industry must dramatically change its attitudes toward safety and regulations. The Commission has recommended that the new regulatory agency prescribe strict standards. At the same time, the Commission recognizes that merely meeting the requirements of a government regulation does not guarantee safety. Therefore, the industry must also set and police its own standards of excellence to ensure the effective management and safe operation of nuclear power plants.

a. The industry should establish a program that specifies appropriate safety standards including those for management, quality assurance, and operating procedures and practices, and that conducts independent evaluations. The recently created Institute of Nuclear Power Operations, or some similar organization, may be an appropriate vehicle for establishing and implementing this program.

b. There must be a systematic gathering, review, and analysis of operating experience at all nuclear power plants coupled with an industry-wide international communications network to facilitate the speedy flow of this information to affected parties. If such experiences indicate the need for modifications in design or operation, such changes should be implemented according to realistic deadlines.

2. Although the Commission considers the responsibility for safety to be with the total organization of the plant, we recommend that each nuclear power plant company have a separate safety group that reports to high-level management. Its assignment would be to evaluate regularly procedures and general plant operations from a safety perspective; to assess quality assurance programs; and to develop continuing safety programs.

3. Integration of management responsibility at all levels must be achieved consistently throughout

this industry. Although there may not be a single optimal management structure for nuclear power plant operation, there must be a single accountable organization with the requisite expertise to take responsibility for the integrated management of the design, construction, operation, and emergency response functions, and the organizational entities that carry them out. Without such demonstrated competence, a power plant operating company should not qualify to receive an operating license.

a. These goals may be obtained at the design stage by (1) contracting for a "turn-key" plant in which the vendor or architect–engineer contracts to supply a fully operational plant and supervises all planning, construction and modification; or (2) assembling expertise capable of integrating the design process. In either case, it is critical that the knowledge and expertise gained during design and construction of the plant be effectively transferred to those responsible for operating the plant.

b. Clearly defined roles and responsibilities for operating procedures and practices must be established to ensure accountability and smooth communication.

c. Since, under our recommendations, accountability for operations during an emergency would rest on the licensee, the licensee must prepare clear procedures defining management roles and responsibilities in the event of a crisis.

4. It is important to attract highly qualified candidates for the positions of senior operator and operator supervisor. Pay scales should be high enough to attract such candidates.

5. Substantially more attention and care must be devoted to the writing, reviewing, and monitoring of plant procedures.

a. The wording of procedures must be clear and concise.

b. The content of procedures must reflect both engineering thinking and operating practicalities.

c. The format of procedures, particularly those that deal with abnormal conditions and emergencies, must be especially clear, including clear diagnostic instructions for identifying the particular abnormal conditions confronting the operators.

d. Management of both utilities and suppliers must insist on the early diagnosis and resolution of safety questions that arise in plant operations. They must also establish deadlines, impose sanctions for the failure to observe such deadlines, and make certain that the results of the diagnoses and any proposed procedural changes based on them are disseminated to those who need to know them.

6. Utility rate-making agencies should recognize that implementation of new safety measures can be inhibited by delay or failure to include the costs of such measures in the utility rate base. The Commission, therefore, recommends that state rate-making agencies give explicit attention to the safety implications of rate making when they consider costs based on "safety-related" changes.

C. Training of Operating Personnel

1. The Commission recommends the establishment of agency-accredited training institutions for operators and immediate supervisors of operators. These institutions should have highly qualified instructors who will maintain high standards, stress understanding of the fundamentals of nuclear power plants and the possible health effects of nuclear power, and who will train operators to respond to emergencies. (See recommendation A.4.a.)

a. These institutions could be national, regional, or specific to individual nuclear steam systems.

b. Reactor operators should be required to graduate from an accredited training institution. Exemption should be made only in cases where there is clear, documentary evidence that the candidate already has the equivalent training.

c. The training institutions should be subject to periodic review and reaccreditation by the restructured NRC.

d. Candidates for the training institute must meet entrance requirements geared to the curriculum.

2. Individual utilities should be responsible for training operators who are graduates of accredited institutions in the specifics of operating a particular plant. These operators should be examined and licensed by the restructured NRC,

both at their initial licensing and at the relicensing stage. To be licensed, operators must pass every portion of the examination. Supervisors of operators, at a minimum, should have the same training as operators.

3. Training should not end when operators are given their licenses.
   a. Comprehensive ongoing training must be given on a regular basis to maintain operators' level of knowledge.
   b. Such training must be continuously integrated with operating experience.
   c. Emphasis must be placed on diagnosing and controlling complex transients and on the fundamental understanding of reactor safety.
   d. Each utility should have ready access to a control room simulator. Operators and supervisors should be required to train regularly on the simulator. The holding of operator licenses should be contingent on performance on the simulator.
4. Research and development should be carried out on improving simulation and simulation systems: (a) to establish and sustain a higher level of realism in the training of operators, including dealing with transients; and (b) to improve the diagnostics and general knowledge of nuclear power plant systems.

D. Technical Assessment

1. Equipment should be reviewed from the point of view of providing information to operators to help them prevent accidents and to cope with accidents when they occur. Included might be instruments that can provide proper warning and diagnostic information, e.g., the measurement of the full range of temperatures within the reactor vessel under normal and abnormal conditions and an indication of the actual position of valves. Computer technology should be used for the clear display for operators and shift supervisors of key measurements relevant to accident conditions, together with diagnostic warnings of conditions.

   In the interim, consideration should be given to requiring, at TMI and similar plants, the grouping of these key measurements, including distinct warning signals, on a single panel available to a specified operator and the providing of a duplicate panel of these key measurements and warnings in the shift supervisor's office.
2. Equipment design and maintenance inadequacies noted at TMI should be reviewed from the point of view of mitigating the consequences of accidents. Inadequacies noted in the following should be corrected: iodine filters, the hydrogen recombiner, the vent gas system, containment isolation, reading of water levels in the containment isolation, reading of water levels in the containment area, radiation monitoring in the containment building, and the capability to take and quickly analyze samples of containment atmosphere and water in various places. (See recommendation A.7.)
3. Monitoring instruments and recording equipment should be provided to record continuously all critical plant measurements and conditions.
4. The Commission recommends that continuing in-depth studies should be initiated on the probabilities and consequences (on-site and off-site) of nuclear power plant accidents, including the consequences of meltdown.
   a. These studies should include a variety of small-break loss-of-coolant accidents and multiple-failure accidents, with particular attention to human failures.
   b. Results of these studies should be used to help plan for recovery and cleanup following a major accident.
   c. From these studies can emerge desirable modifications in the design of plants that will help prevent accidents and mitigate their consequences. For example:
      (i) Consideration should be given to equipment that would facilitate the controlled safe venting of hydrogen gas from the reactor cooling system.
      (ii) Consideration should be given to overall gas-tight enclosure of the let-down/make-up system with the option of returning gases to the containment building.
   d. Such studies should be conducted by the industry and other qualified organizations and may be sponsored by the restructured NRC and other federal agencies.
5. A study should be made of the chemical behavior and the extensive retention of radioactive iodine in water, which resulted in the very low release of radioiodine to the atmosphere in the TMI-2 accident. This information should be taken into account in the studies of the consequences of other small-break accidents.

6. Since there are still health hazards associated with the cleanup and disposal process, which is being carried out for the first time in a commercial nuclear power plant, the Commission recommends close monitoring of the cleanup process at TMI and of the transportation and disposal of the large amount of radioactive material. As much data as possible should be preserved and recorded about the conditions within the containment building so that these can be used for future safety analyses.
7. The Commission recommends that, as a part of the formal safety assurance program, every accident or every new abnormal event be carefully screened, and where appropriate be rigorously investigated, to assess its implications for the existing system design, computer models of the system, equipment design and quality, operations, operator training, operator training simulators, plant procedures, safety systems, emergency measures, management, and regulatory requirements.

E. Worker and Public Health and Safety

1. The Commission recommends the establishment of expanded and better coordinated health-related radiation effects research. This research should include, but not be limited to:
   a. Biological effects of low levels of ionizing radiation.
   b. Acceptable levels of exposure to ionizing radiation for the general population and for workers.
   c. Development of methods of monitoring and surveillance, including epidemiologic surveillance to monitor and determine the consequences of exposure to radiation of various population groups, including workers.
   d. Development of approaches to mitigate adverse health effects of ionizing radiation.
   e. Genetic or environmental factors that predispose individuals to increased susceptibility to adverse effects.

This effort should be coordinated under the National Institutes of Health—with an interagency committee of relevant federal agencies to establish the agenda for research efforts—including the commitment of a portion of the research budget to meet the specific needs of the restructured NRC.

2. To ensure the best available review of radiation-related health issues, including reactor siting issues, policy statements or regulations in that area of the restructured NRC should be subject to mandatory review and comment by the Secretary of the Department of Human Services. A time limit for the review should be established to assure such review is performed in an expeditious manner.
3. The Commission recommends, as a state and local responsibility, an increased program for educating health professionals and emergency response personnel in the vicinity of nuclear power plants.
4. Utilities must make sufficient advance preparation for the mitigation of emergencies:
   a. Radiation monitors should be available for monitoring of routine operations as well as accident levels.
   b. The emergency control center for health-physics operations and the analytical laboratory to be used in emergencies should be located in a well-shielded area supplied with uncontaminated air.
   c. There must be a sufficient health-related supply of instruments, respirators, and other necessary equipment for both routine and emergency conditions.
   d. There should be an adequate maintenance program for all such health-related equipment.
5. An adequate supply of the radiation protection (thyroid blocking) agent, potassium iodide for human use, should be available regionally for distribution to the general population and workers affected by a radiological emergency.

F. Emergency Planning and Response

1. Emergency plans must detail clearly and consistently the actions public officials and utilities should take in the event of off-site radiation doses resulting from release of radioactivity. Therefore the Commission recommends that:
   a. Before a utility is granted an operating license for a new nuclear power plant, the state within which that plant is to be sited must have an emergency response plan reviewed and approved by the Federal Emergency Measurement Agency (FEMA). The agency should assess the criteria and procedures now used for evaluating state and local government plans and for determining their ability to activate the plans. Adequate provisions must

be assured by FEMA, where necessary, for multistate planning.

b. The responsibility at the federal level for radiological emergency planning, including planning for coping with radiological releases, should rest with FEMA. In this process, FEMA should consult with other agencies, including the restructured NRC and the appropriate health and environmental agencies. (See recommendation A.4.)

c. The state must effectively coordinate its planning with the utility and with local officials in the area where the plant is to be located.

d. States with plants already operating must upgrade their plans to the requirements to be set by FEMA. Strict deadlines must be established to accomplish this goal.

2. Plans for protecting the public in the event of off-site radiation releases should be based on technical assessment of various classes of accidents that can take place at a given plant.

a. No single plan that is based on a fixed set of distances and a fixed set of responses can be adequate. Planning should involve the identification of several different kinds of accidents with different possible radiation consequences. For each such scenario there should be clearly identified criteria for the appropriate responses at various distances, including instructing individuals to stay indoors for a period of time, providing special medication, or ordering an evacuation.

b. Similarly, response plans should be keyed to various possible scenarios and activated when the nature and potential hazard of a given accident has been identified.

c. Plans should exist for protecting the public at radiation levels lower than those currently used in NRC-prescribed plans.

d. All local communities should have funds and technical support adequate for preparing the kinds of plans described above.

3. Research should be expanded on medical means of protecting the public against various levels and types of radiation. This research should include exploration of appropriate medications that can protect against or counteract radiation.

4. If emergency planning and response to a radiation-related emergency is to be effective, the public must be better informed about nuclear power. The Commission recommends a program to educate the public on how nuclear power plants operate, on radiation and its health effects, and on protection actions against radiation. Those who would be affected by such emergency planning must have clear information on actions they would be required to take in an emergency.

5. Commission studies suggest that decision makers may have overestimated the human costs, in injury and loss of life, in many mass evacuation situations. The Commission recommends study into the human costs of radiation-related mass evacuation and the extent, if any, to which the risks in radiation-related evacuation differ from other types of evacuations. Such studies should take into account the effects of improving emergency planning, public awareness of such planning, and costs involved in mass evacuations.

6. Plans for providing federal technical support, such as radiological monitoring, should clearly specify the responsibilities of the various support agencies and the procedures by which those agencies provide assistance. Existing plans for the provision of federal assistance, particularly the Interagency Radiological Assistance Plan and the various memoranda of understanding among the agencies, should be reexamined and revised by the appropriate federal authorities in the light of the experience of the TMI accident, to provide for better coordination and more efficient federal support capability.

G. The Public's Right to Information

1. Federal and state agencies, as well as the utility, should make adequate preparation for a systematic public information program so that in time of a radiation-related emergency, they can provide timely and accurate information to the news media and the public in a form that is understandable. There should be sufficient division of briefing responsibilities as well as availability of informed sources to reduce confused and inaccurate information. The Commission therefore recommends:

a. Since the utility must be responsible for the management of the accident, it should also be primarily responsible for providing information on the status of the plant to the news media and to the public; but the restructured NRC should also play a supporting role and be available to provide background information and technical briefings.

b. Since the state government is responsible for decisions concerning protective actions, including evacuations, a designated state agency should be charged with issuing all information on this subject. This agency is also charged with the development of and dissemination of accurate and timely information on off-site radiation doses resulting from releases of radioactivity. This information should be derived from appropriate sources. (See recommendation F.1.) This agency should also set up the machinery to keep local officials fully informed of developments and to coordinate briefings to discuss any federal involvement in evacuation matters.

2. The provision of accurate and timely information places special responsibilities on the official sources of this information. The effort must meet the needs of the news media for information but without compromising the ability of operational personnel to manage the accident. The Commission therefore recommends that:
   a. Those who brief the news media must have direct access to informed sources of information.
   b. Technical liaison people should be designated to inform the briefers and to serve as a resource for the news media.
   c. The primary official news sources should have plans for the prompt establishment of press centers reasonably close to the site. These must be properly equipped, have appropriate visual aids and reference materials, and be staffed with individuals who are knowledgeable in dealing with the news media. These press centers must be operational promptly upon the declaration of a general emergency or its equivalent.

3. The coverage of nuclear emergencies places special responsibilities on the news media to provide accurate and timely information. The Commission therefore recommends that:
   a. All major media outlets (wire services, broadcast networks, news magazines, and metropolitan daily newspapers) hire and train specialists who have more than a passing familiarity with reactors and the language of radiation. All other news media, regardless of their size, located near nuclear power plants should attempt to acquire similar knowledge or make plans to secure it during an emergency.
   b. Reporters discipline themselves to place complex information in a context that is understandable to the public and that allows members of the public to make decisions regarding their health and safety.
   c. Reporters educate themselves to understand the pitfalls in interpreting answers to "what if" questions. Those covering an accident should have the ability to understand uncertainties expressed by sources of information and probabilities assigned to various possible dangers.

4. State emergency plans should include provision for creation of local broadcast media networks for emergencies that will supply timely and accurate information. Arrangements should be made to make available knowledgeable briefers to go on the air to clear up rumors and explain conditions at that plant. Communications between state officials, the utility, and the network should be prearranged to handle the possibility of an evacuation announcement.

5. The Commission recommends that the public in the vicinity of a nuclear power plant be routinely informed of local radiation measurements that depart appreciably from normal background radiation, whether from normal or abnormal operation of the nuclear power plant, from a radioactivity cleanup operation, such as that at TMI-2, or from other sources.

ANEXO 5

Operating Experiences

Edited by William R. Casto

The Rogovin Report on Three Mile Island 2

[Editor's Note: The last of several major inquiries into the Mar. 28, 1979, accident at Unit 2 of the Three Mile Island nuclear plant was the special inquiry sponsored by the Nuclear Regulatory Commission (NRC) but conducted by the law firm of Rogovin, Stern & Huge, from which the report derives its name. The report, released in late January 1980, is entitled *Volume 1, Three Mile Island: A Report to the Commissioners and to the Public.* There is no report number, but the volume is available from NRC and NTIS.

Presented here are the "Foreword" and the "Summary" from the Rogovin report.

FOREWORD

Within weeks of the Mar. 28, 1979, accident at Three Mile Island Unit 2 (TMI-2), the Nuclear Regulatory Commission (NRC) decided to institute a special inquiry to review and report on the accident. The principal objectives of the inquiry were to determine what happened and why, to assess the actions of utility and NRC personnel before and during the accident, and to identify deficiencies in the system and areas where further investigation might be warranted.

The work of the special inquiry was not intended to duplicate the efforts of the President's Commission on The Accident at Three Mile Island. The inquiry was designed to enable the NRC to fulfill its regulatory responsibilities by achieving the fullest possible understanding of the accident, both from a technical point of view and from the standpoint of how the NRC's own regulatory processes functioned.

Recognizing the potential conflict-of-interest problems involved if the inquiry were directed and undertaken solely by the NRC staff, the NRC in mid-June 1979 contracted with our law firm, Rogovin, Stern & Huge, to conduct the inquiry, specifying that it would have full independence in carrying out the work. Neither the law firm nor any of its members had any prior involvement with nuclear energy issues.

The bulk of the inquiry staff consisted of volunteer NRC professionals chosen for their expertise in subjects to be pursued by the inquiry. Some of these individuals had been with the NRC and its predecessor, the Atomic Energy Commission, for many years; others were relative newcomers to the NRC from congressional staffs or academia. To ensure the inquiry's competence and independence, we reviewed the NRC staff members' professional qualifications and any prior involvement they had had in the accident or the licensing of the two reactors at Three Mile Island so that we could be alert to any possible conflicts of interest.

In addition to NRC staff members, a number of technical consultants in the areas of accident investigation and safety management, and lawyers experienced in the conduct of investigations were hired to assist the inquiry full time. Work was also performed for the inquiry in several technical or specialized areas by national laboratories, by the National Academy of Public Administration (to convene an expert panel on emergency response), and by a firm expert in human factors engineering.

During the investigative stage of the inquiry, we conducted approximately 270 formal depositions under oath. Those deposed included the five NRC Commissioners, dozens of top NRC staff officials, the management of both the utility company that operates Three Mile Island and the reactor's manufacturer, and the control-room crews. In addition, we had access to the transcripts of interviews and depositions taken by other investigations, including the President's Commission and NRC's own enforcement investigation, and conducted hundreds of additional interviews.

In a further effort to improve the quality of our inquiry and to ensure that we would not be "captured" by the staff, the personal views of some 21 outside consultants expert in the field of nuclear safety were sought on the intended course of the inquiry and its final product. These consultants, from universities, national laboratories, industry, and the public interest sector, were chosen not only for their knowledge and judgment but also for their wide spectrum of views. They met with us in teams of three to six early in the inquiry to examine our work plans and again near the end of the project to comment on draft sections of our final report; each responded in detail to a draft of our proposed conclusions and recommendations. Our understanding with these consultants from the beginning was that they would not be solicited to "endorse" our conclusions, but rather to educate us and to criticize our effort in order to help us improve it.

Although our inquiry was specifically focused on the accident at TMI-2, we were asked to reach conclusions and to make recommendations with a broader sweep than the accident itself. Since "extrapolation" has been recognized to be "the fertile father of error," we wish to point out at the outset that, where we do make generalizations about the industry as a whole, there are notable exceptions. We hope the reader will bear this in mind.

In any project such as this special inquiry, where the combined talents of some 70 nuclear engineers, scientists, lawyers, and investigators have been blended together with a large number of outside consultants, it is most important to make clear where responsibility lies for the conclusions and recommendations found in these volumes.

We have had the benefit of a superb technical staff. They have unstintingly given us an unmatched level of excellence, as have our 21 individual consultants. Without their hard work and the quality of their contributions—indeed, without the unique interplay of diverse viewpoints to which we have had the good fortune to be exposed during this inquiry—this report would not have been possible. Although all these talented people have guided us through the rocks and shoals of an extraordinarily technical inquiry, in the end we must take sole responsibility for the report—and particularly for the conclusions and recommendations, which are ours alone. We hope they are objective, informed, and clear.

Finally, a note about the organization of this report. For years, the nuclear energy debate was carried on in this country by a relative handful of people. Three Mile Island has changed that. The nuclear power issue has become riveted to the American consciousness. Volume I of this report contains a narrative account of the accident and a discussion of our overall conclusions and recommendations. Volume II contains more detailed factual and technical material, from which Volume I is drawn in large part.

The narrative account of the accident in Volume I is written in an open, nontechnical style in the hope that it will be widely read and will encourage the reader to examine the conclusions and recommendations that comprise the second portion of this volume.

SUMMARY

The one theme that runs through the conclusions we have reached is that the principal deficiencies in commercial reactor safety today are not hardware problems, they are management problems. These problems cannot be solved by the addition of a few pipes and valves—or, for that matter, by a resident federal inspector at every reactor. Undoubtedly improvements in the design, instrumentation, and control logic of nuclear plants can be made to reduce the probability of a serious accident and to better protect the public should such an accident occur. Some detailed suggestions for such improvements are included here and in the in-depth studies to this report. But the most serious problems will be solved *only* by fundamental changes in the industry and the NRC.

What we have found is a regulatory system consisting primarily of an elaborate apparatus for reviewing the safety of nuclear reactor designs which has served the public well in the past and produced a good safety record to date but in the process has failed to take timely account of the actual operation of existing plants. We have found that the NRC is not focused, organized, or managed to meet today's needs. In our opinion the NRC is incapable, in its present configuration, of managing a comprehensive national safety program for existing nuclear power plants and plants scheduled to come on-line in the next few years which is adequate to ensure the public health and safety.

We have found, based on our study of TMI-2 and our interviews with knowledgeable people in the industry, that many nuclear plants are probably operated by management that has failed to make certain that enough properly trained operators and qualified engineers are available on site in responsible positions to diagnose and cope with a potentially serious accident. The NRC, for its part, has virtually ignored the critical areas of operator training, human factors engineering, utility management, and technical qualifications.

We have found an industry in which the expertise and responsibility for safety is fragmented among many parties—the utility company that operates the plant, the plant designer, the manufacturer of the reactor system, the contractor, and the suppliers of critical components, in addition to the NRC. Coordination among these parties and between them and the NRC, as well as within the NRC, is inadequate. As a result, there are many institutional disincentives to safety, and safety issues that are identified at some point in the system often fall through the cracks. Prior to TMI-2, the industry as a whole had made only feeble attempts to mount any industrywide affirmative safety program, and many utilities apparently regarded bare compliance with NRC minimum regulations as more than adequate for safety.

On top of all this, we found that before Mar. 28, 1979, an attitude of complacency pervaded both the industry and the NRC, an attitude that the engineered design safeguards built into today's plants were more than adequate, that an accident like that at TMI-2 would not occur—in the peculiar jargon of the industry, that such an accident was not a "credible event."

The kinds of changes needed to cope with these problems and attitudes are institutional, organizational, and managerial, and include:

—An immediate, substantial shift in the balance of existing resources in the NRC from design review to the monitoring of operating reactors and consolidation of these resources in one NRC office; new mechanisms to evaluate operating experience and to ensure that necessary changes are implemented in the regulatory program; and an improved inspection and evaluation system for operating reactors.

—Strong measures to strengthen the on-site technical capability and management of utilities at reactor sites, including a new philosophy and new programs for improved operator training; and new NRC requirements to ensure that qualified engineer supervisors with intimate knowledge of the plant will be part of the on-site supervisory management chain on every reactor operating shift.

—The chartering of an operating consortium with the capability to operate the plants of a number of utilities on either a contract or "receivership" basis.

—For future reactors, more remote siting. For existing reactors, the promulgation by the NRC of specific criteria for determining the minimum evacuation planning zone around each plant; the conditioning of operating licenses on such plans being approved and workable; and the closing down of existing plants that cannot meet these new criteria, unless either (1) additional safety systems for mitigation of accidents are installed or (2) the President determines that the continued operation of the plant is vital to the national interest.

—In the case of new applications for reactor licenses, a completely overhauled licensing system that includes one-stage licensing; increased standardization; increased use of rule-making proceedings by the NRC to implement safety policy and standards; establishment of an Office of Public Counsel; and agency funding of intervenors who make material, substantive contributions to licensing and rule-making proceedings.

—Substantial changes in the bases used to review the safety of reactor designs, including the application of quantitative risk-assessment methods to potential accident sequences in order to augment the current "design-basis-accident" approach.

The accident at TMI-2 did not result in radioactive release levels that posed any threat to public health, even in the long run. Public alarm over radioactivity fueled by the Governor's evacuation advisory to pregnant women and preschool children 2 days after the accident, and the fear caused by reports the next day and afterwards of a possible hydrogen bubble explosion, turn out to have been vastly exaggerated by the NRC's disorganized response to the emergency.

But engineering calculations performed during our investigation indicate that on the morning of March 28, before anyone appreciated the seriousness of the situation, TMI-2 came close to being the accident we had been told by many in the industry could not happen: a core meltdown. A shift foreman who reported for normal duty about 2 hr after the accident began undertook to survey some instruments and blocked off the stuck-open pressurizer valve that was leaking reactor coolant into the reactor containment building. If that block valve had remained open, our projections show that within 30 to 60 min a substantial amount of the reactor fuel would have begun to melt down—requiring at least the precautionary evacuation of thousands of people living near the plant and potentially serious public health and safety consequences for the immediate area.

An accident identical to that at TMI-2 is not going to happen again. Changes have been made to ameliorate the particular problems revealed there, and the accident has spawned major reexamination by the industry and the NRC of many aspects of design and operations that contributed to that accident.

However, the work done by the Special Inquiry Group over the past 7 months has led us to conclude that unless *fundamental* changes, such as those outlined above, are made in the way commercial nuclear reactors are built, operated, and regulated in this country, similar accidents—perhaps with potentially serious consequences to the public health and safety which were only narrowly averted at TMI-2—are likely to recur.

We were not asked, and it is not our place to tell the public, "how safe is safe enough." Indeed, as we make clear in this report, we believe this is a decision that in the final analysis should not be the exclusive province of the NRC; it is an executive decision that should be made as a part of our national energy strategy by the Executive and by Congress. The NRC cannot continue to face sub silentio in every policy and licensing determination the question of the future of nuclear power in this country. It is, lest we forget, an inherently dangerous activity that Congress has authorized the NRC to license.

The generation of nuclear power can never be risk-free. It will inevitably present certain risks to the public health and safety no matter how "safe" plants are made. Available surveys show, however, that nuclear power is unique in that public *perception* of the risk of injury or health consequences—even the risk involved once an accident occurs—is many, many times greater than the best available estimates of *actual* risk—far more so than for any other potentially harmful everyday or industrial activity. Just as the regulators must change their attitudes to appreciate that this perception of risk cannot be dealt with by trying to convince the public that it "cannot happen," so renewed efforts must be made to educate the public that the risks and benefits associated with nuclear power plants must be weighed against the very real health and environmental risks associated with other forms of power generation, such as the increased use of coal and synthetic fuels, and against such risks as continued dependence on foreign oil imports.

We considered at great length and rejected a recommendation that there be a moratorium on operating reactors or on granting new operating licenses for reactors now under construction and nearing completion. We do believe, however, that the NRC's management would be wise to suspend processing of applications for construction permits and limited work authorizations until it considers the various recommendations that we have made for reforming the licensing process and for increased standardization.

In an investigation like this, the very purpose of which is to focus on what went wrong and what needs changing, it is inevitable that less attention than is deserved will be given to what "went right"—the strong points in the system. Chief among these is the fact that the "defense-in-depth" concept worked to protect the public health and safety. In spite of multiple equipment malfunctions, human failures, and the creation of conditions in the reactor and auxiliary buildings that were never contemplated in the design of the plant's safety systems, the utility and its engineering support staff were able to bring the system to a stable condition without releases of radioactive materials to the atmosphere that could have resulted in significant health effects to those living near the plant.

Thereafter a massive response by industry, the national laboratories, and the government greatly assisted the Metropolitan Edison Company both in bringing the plant to a safe shutdown and in establishing recovery operations. Over a thousand people, from reactor operators and health-physics technicians to top executives from every corner of the industry, dropped their everyday work and went to the TMI site. Thousands more were active in performing supporting analyses and experiments and in procuring and dispatching needed supplies.

That there are strengths in the system does not, however, detract from the urgent need to make changes where important weaknesses have been revealed. These changes will not be easy; they will require new legislation, executive reorganization, and substantial overhaul of the way the NRC is organized and managed, at the very least. But the changes are feasible, and they will not require such vast expenditures of money or other resources as to be beyond the bounds of reason.

What the changes require is a firm commitment on the part of the President and the congressional oversight committees each to play its own role and a commitment by the public—if what it wants is *safer* nuclear power plants—to keep the pressure on its elected representatives for major, meaningful reform. For in the polarization of the current public debate over nuclear power, we have found there is precious little constituency for that course. On the one side are those who do not want reforms and do not want to put any more resources into nuclear power at all because they believe it should be shut down; on the other side are those who argue that existing plants and the program for operating and regulating them adequately protect the public and that major reforms are not necessary.

What we *are* able to conclude confidently, based on the work of the Special Inquiry Group, is that, although the changes that must be made are major ones, these changes *will* make commercial nuclear power much safer than it is today. In our view, if a firm commitment is not made promptly to bring about these changes, we will be exposing the public to a needlessly high level of risk.

Finally, we believe a summary comment is called for with respect to the recommendations made by this and other inquiries. Over the years the nuclear industry and its regulators have identified what have been considered to be serious safety problems and recommendations whose significance has been underscored by ringing statements to the effect that unless such problems are resolved "promptly," a license should be revoked or the industry shut down. Many of these problems are still outstanding. Although we do not undertake to set out deadlines, we do believe that the congressional oversight committees should hold the NRC accountable with respect to such issues.

The Rogovin report continues with a lengthy discussion of its conclusions and recommendations in each of the following 12 areas:

1. Systematic Evaluation of Operating Experience and Improvement in the Regulation of Operating Reactors.
2. Strengthening the On-Site Technical and Management Capability of the Utility: Improved Operator Training and New NRC Requirements for Qualified Engineer Supervisors on Every Shift.
3. Chartering of a National Operating Company or Consortium.
4. Improved NRC Management and Reorganization to a Single Administrator Agency; Establishment of an Independent Reactor Safety Board.
5. Greater Application of Human Factors Engineering, Including Better Instrumentation Display and Improved Control-Room Design.
6. More Remote Siting and Improved Emergency Planning, Including Workable Evacuation Planning as a Condition of Reactor Operation.
7. Overhaul of the Licensing Process: One-Stage Licensing Increased Standardization, Increased Use of Rule Making, Establishment of an Office of Public Counsel, and Intervenor Funding.
8. Improvement in the Basis for Safety Review of Reactor Design and Increased Use of Quantitative Risk-Assessment Techniques.
9. Health Effects from Radioactive Releases During the Accident and Occupational Health Physics at the Site.
10. Information Made Available to the News Media.
11. Sabotage, Bribery, and Cover-up.
12. Disincentives to Safety.

ANEXO 6

Selected Safety-Related Events Reported in March and April 1980

Compiled by Wm. R. Casto

Of the incidents reported during this period, two are reviewed because of their uniqueness and general interest: (1) at Prairie Island 1 a steam-generator tube ruptured, which must have sounded familiar to designers; and (2) Crystal River 3 experienced an instrumentation and control failure that resulted in the ejection of about 150 000 liters of reactor coolant into the containment.

6.1 STEAM-GENERATOR TUBE RUPTURE AT PRAIRIE ISLAND

On Oct. 2, 1979, a tube break occurred in steam generator (SG) No. 11 of **Prairie Island 1**, which is owned by the **Northern States Power Company.**[1] All engineered safety systems functioned as designed, and the plant operating staff shut the reactor down, isolated the SG, and cooled the reactor coolant system (RCS), following existing operating procedures.

Operational Sequence of Events

The following is a chronological sequence of events that occurred from the first indication of a break in an SG tube until the leaking tube was identified.

Date and time	Event
October 2	
1414	High-radiation alarm on air-ejector discharge gaseous radiation monitor
1420	Overtemperature ΔT turbine runback signal due to decreasing pressure (no actual runback occurred; high leakage started)
1421	Low pressurizer pressure [14.7 MPa (<2140 psig)]
~1421	Commenced load reduction
1422	Low pressurizer level (<18.3%)
1423	Started second charging pump (No. 11)
~1424	Started third charging pump (No. 13)
1424:09	Reactor trip for "low pressurizer pressure" [13.3 MPa (<1900 psig)]
1424:14	Safety injection (SI) occurred owing to "low pressurizer pressure" [12.7 MPa (<1815 psig)]
October 2 (Continued)	
1424:33	Minimum RCS water inventory; RCS pressure begins increasing
1426	Reactor coolant pump No. 11 stopped
1427	Reactor coolant pump No. 12 stopped
1432:29	SG No. 11 level increased above the "lo lo level" set point (13%) on the narrow range after having gone off scale low after the trip (it is normal for the SG level to go off scale low on a trip; recovery in this case was much more rapid than usual)
>1438	SI system reset
1441	Loop A main steam isolation valve (MSIV) closed
1456	Pressurizer level returned on scale
1456	Stopped SI pump No. 12
1456–1457	Began depressurization of the RCS using the pressurizer power-operated relief valve (the valve was cycled six to eight times to reduce pressure to required value)
1502	Pressurizer level reached the high-level set point (>55%)
1506	SI pump No. 11 stopped
1507	Pressurizer-relief-tank rupture disk ruptured
1515	RCS pressure at 6.4 MPa (910 psig), same as SG No. 11 pressure; leak apparently stopped
1550	Commenced normal cooldown
October 3	
0640	Residual heat-removal system placed in service to continue cooling to cold shutdown
1300	RCS at cold shutdown
October 6	
1640	Completed draining of RCS
October 7	
0250	Identified leaking tube

Several points relating to cooldown need to be considered during recovery from an event such as this:

1. Since the leaking SG cannot be used for cooling, RCS cooldown will tend to be slower. Also, there will be some delay in cooling the leaking SG to the point where entry is possible.

2. With a leak next to the tube sheet, some SG water will drain into the RCS; therefore adequate boron must be injected. In this case a second boric acid tank was used to raise the boric acid concentration. In SG No. 11 a sufficiently high boron concentration was measured to ensure no problems with RCS dilution during the cooldown.

3. Since large quantities of 12% boric acid were used for injection via the SI pumps to the cold legs, these lines must be flushed.

Operational Parameter Behavior

The Prairie Island plant computer has the "posttrip review" option, which accumulates key parameter data on a continuing basis and saves the data immediately preceding and after a reactor trip. In addition, software that plots the following data has been developed: pressurizer level, pressurizer pressure, nuclear power, $T_{ave.}$ (average temperature), T_c (temperature of the cold leg), ΔT, and steam pressure (secondary).

The pressurizer level went off scale low before the safety injection (SI) occurred at 12.7 MPa (1815 psig). Accident analyses in the past had assumed initiation of the SI signal due to the coincident low pressurizer level (5%) and low pressurizer pressure [12.7 MPa (1815 psig)] signals. In May 1979 the SI actuation scheme had been modified, as a result of Three Mile Island follow-up evaluations, to a 2-out-of-3 coincidence, low-pressurizer-pressure actuation logic. With either scheme, SI signal initiation would have had to wait until pressure had dropped below 12.7 MPa (1815 psig). Thus the safety analysis still bounds the plant response.

Pressure during the most severe portion of the transient was dropping at a rate of ~0.7 MPa (100 psi/min), but pressure recovery was rapid once the SI pumps started. The plots of nuclear power and temperature differential (ΔT) show the power reduction before the trip, and $T_{ave.}$ increased as expected to accompany the load drop since the control rods were at the withdrawal limit. The temperature of the cold leg (T_c) also increased as expected after the load reduction and was leveling out at ~278°C when the trip occurred.

Training for Natural-Circulation Cooldown

After the reactor trip and the safety injection, the reactor coolant pumps were tripped in accordance with Office of Inspection and Enforcement Bulletin 79-06C. Cooldown was accomplished using natural circulation until 2128 on Oct. 2, when reactor coolant pump No. 12 was restarted to allow cooldown using normal procedures. Natural-circulation tests and demonstrations have been conducted as part of the preoperational startup testing program and operator requalification training exercises. In fact, most operators have had an opportunity to cool down the plant using natural circulation. Thus the operators were aware of what to expect in regard to parameter behavior. Cooldown rates on the order of 10°C/h (18°F/h) were obtained with a ΔT of ~17°C (30°F). Cooldown was accomplished by directing steam dump to the main condenser during the first phase of the cooldown.

The Prairie Island requalification program for licensed operators normally includes simulator training in emergency procedures and accident simulation. Exercises in which SG tube ruptures, as well as a variety of other accident conditions, are simulated have typically been included in these training programs.

The maximum leak rate was calculated to be 1476 liters/min. This calculation is based on pressurizer level vs. time from the posttrip review data.

The pressurizer level is calibrated so that 0% should correspond to the lower-level tap on the pressurizer for hot pressurizer conditions. For a different set of conditions, the lower-level tap may correspond to a slightly positive or a slightly negative reading. In this case it corresponded to a reading of about −5%, the point at which the pressurizer level indication became flat. A conservative calculation indicated that the lower tap was uncovered for ~86 s. The volume at this point would be 1189 liters. It was further calculated that 1033 liters was lost from the pressurizer after the lower tap was uncovered. Therefore the minimum calculated inventory was 155 liters.

Radioactive Releases

Gas releases were made through the following paths: air-ejector discharge, turbine-driven auxiliary feed-pump exhaust, atmospheric steam dump, and gland steam exhaust.

Periodic grab samples were taken of the air-ejector exhaust to accurately assess the radioactivity it released. The air ejector was used from 1414, Oct. 2, 1979, to 0730, Oct. 3, 1979. Even though SG No. 11 was isolated at 1441, the health physicists deemed it prudent to monitor the exhaust until the unit was secured. Approximately 27 Ci of gaseous radioactivity, most of which was ^{133}Xe (21.9 Ci) and ^{135}Xe (4.63 Ci), was released via the air ejector during this period.

The turbine-driven auxiliary feed pump was operated for ~24 min (until after SI was reset). On the basis of the SG activity data at 1515 and 1715 and assuming full flow with the auxiliary feed pump (a conservative assumption), the radioactive releases from the turbine of this steam-driven pump were <0.5 Ci of gaseous radioactivity (mostly ^{133}Xe and ^{135}Xe) and 40 μCi of iodine, mostly ^{131}I.

One of the two atmospheric steam-dump valves on loop A opened for 3 s after the reactor trip. On the basis of previous operating experience, this time is fairly typical. These valves normally take about 3 s to stroke fully open. It was assumed that full flow passed for 3 s in order to determine the possible release via this path, and radioactivity was back calculated from samples at 1515 and 1715 (as for the auxiliary feed-pump evaluation). The calculated releases via the steam dump were about one-tenth of that from the auxiliary feed pump.

For all these release paths, the iodine releases were based only on data at 1515, which are more realistic. Other gaseous releases were conservatively calculated by using the data at 1715 since more complete data were available. No estimate of the release via the gland steam exhaust is available at this time. Release by this path is expected to be less than that from the air ejector.

Liquid Radioactivity

The air-ejector condenser drains to the turbine-building sump. Liquid samples were taken when releases were made from the sump. From Oct. 2 to Oct. 4, 1979, 0.023 μCi of ^{133}Xe and ^{135}Xe was released from the turbine-building sump.

Steam-generator blowdown was being directed to the condenser before the tube failure; this eliminated one possible path for liquid release to the environment. The circulating-water system was operating in the recycle mode in which only 4250 liters/s is returned to the Mississippi River. This tended to spread out the releases from the turbine-building sump to the river over a longer period. There was no detectable iodine in the liquid samples during this period.

Off-Site Thermoluminescent-Detector Measurements

Thermoluminescent detectors (TLDs) had been placed in four locations near the Prairie Island facility on Oct. 2, 1979. After an exposure of ~28 h, which covered the time of the incident, they were removed and analyzed. The readings were between 0.18 and 0.33 mrem ± 0.40.

Behavior of Iodine in the Reactor Cooling System

Over 90 ^{131}I samples were taken from the RCS and measured in the 5½ d after the event because of the interest in the spiking phenomenon and the transfer of the coolant from the RCS to the environment. The ^{131}I activity varied from $\sim 2.5 \times 10^{-2}$ μCi/ml preceding the reactor trip to a peak of $\sim 5.6 \times 10^{-2}$ μCi/ml at ~2000 on Oct. 2, then leveled out at $\sim 8.5 \times 10^{-3}$ μCi/ml after Oct. 3. The leveling out was attributed to the fact that the purification system ion exchangers had been valved out after the trip to prevent reduction in the RCS boron concentration.

It should be noted that all actual concentrations of radionuclides following this incident were well below the postulated concentration in the facility's Final Safety-Analysis Report (FSAR).

Steam-Generator Inspection and Corrective Actions

Evaluating the break and identifying the tube that failed in the steam generator was accomplished by draining water from the secondary side of the generator into the RCS through the opening of the leaking tube. Once the water on the secondary side stopped draining, the break elevation was determined. Then, by slowly adding water to the secondary side and visually inspecting the tube sheet from the primary side through the manways, the specific tube was identified.

Since the leaking tube was located in the outer periphery of the tube bundle (inlet side) and the break was just above the tube sheet within the flow lane, it was suspected that the damage was caused by a foreign object. As a result of the initial findings, an eddy-current inspection was concentrated on the outer-periphery tubes. The results of this examination revealed that one tube (adjacent to the tube that failed) had an indication of ~65% reduction in wall thickness and that an adjacent tube had ~20% reduction in wall thickness. All indications were at approximately the same elevation.

With the aid of mirrors and fiber optics, a visual examination of the three degraded tubes verified the eddy-current results and revealed that the break resembled a classical overpressure burst (running ~38 mm in the longitudinal direction of the tube with an opening width of ~13 mm). The other two tubes showed signs of wear. All wear marks were on the outer peripheral side of the tubes.

During this time a coil spring—later measured to be 216 mm long, 32 mm in diameter, and $\frac{3}{32}$ gauge—was found on the tube sheet adjacent to the defective tubes. One end of the spring was lodged under the tube-lane blocking device, and the other end was free to move. A definite wear pattern on the tube sheet indicated that the spring had moved during operation. Later, a second spring, identical to the first, was found on the cold-leg side. It was located just opposite the first spring in a similar condition, with one end lodged under the tube-lane blocking device and the other end extending out onto the tube sheet. In addition, part of an aviation hose clamp was found next to this spring on the cold-leg side. A close visual examination of the spring, the tubes, and the tube sheet of the cold-leg side revealed no signs of spring movement, tube damage, or wear. This was confirmed by eddy-current examination of all eight tubes in close proximity to the second spring.

Once these objects were removed from the steam generator, it was apparent that the spring and piece of hose clamp from the cold-leg side were encrusted with oxides, indicating no active movement, whereas the coils of the spring from the hot-leg side showed definite signs of wear.

The springs and clamp appeared to have been part of sludge-lancing equipment used in one of the previous outages, and it appears that this spring was dropped into the steam generator before the tube-lane blocking device was installed. It was concluded that the tube break was the result of reduction of wall thickness by the wearing action of the spring against the tube.

In addition to the failed tube and the tube with 65% wall reduction, the remaining four tubes adjacent to the failed tube were plugged.

Comparison of FSAR Results vs. Results of an Actual Break

The Prairie Island FSAR, Sec. 14.2.4, addresses an SG tube rupture. Table 1 summarizes the differences between the FSAR calculations and the results of the actual break on October 2.

The FSAR assumed that specific radionuclide concentrations in the RCS coolant were 2 orders of magnitude less than the actual concentrations in the RCS coolant as measured at 0730 on Oct. 2, 1979, except for ^{54}Mn (a corrosion product) which was only a factor of 3 less.

Table 1 Comparison of FSAR and Results of Actual Break in an SG Tube at Prairie Island

Item	FSAR	Prairie Island tube break
Leak rate, liters/min	~2330	~1480
Defective fuel, %	1.0	~0.01
Previous leak rate (before break), liters/min	19	0.0
Steam transferred in 30 min, mg	54.4	2.3*
Radioactivity released, Ci		
^{133}Xe equivalent	21 700	30 (actual ^{133}Xe)
^{131}I	209	37.0×10^{-6} (dose equivalent)

*Steam was released from operation of the turbine-driven auxiliary feed pump and from actuation of the atmosphere steam dump system to atmosphere actuation. Actual water transferred was significantly more.

Recommendations

A review of the event has resulted in the following recommendations and comments:

1. A note of caution should be added to the SG tube-rupture procedure: the operator should stop the turbine-driven auxiliary feed pump as soon as possible and shut the steam supply valve from the affected steam generator. This was done even though this cautionary note was not included in the procedure.

2. Add a note of emphasis to the operator to isolate the leaking steam generator as soon as possible and to keep in mind that the MSIV bypass can be used to protect the steam generator from overpressure. Also, reducing the RCS pressure quickly while maintaining adequate RCS subcooling, which was done during this event, will help prevent overpressure of the secondary side.

3. Consider operation of the reactor coolant pumps during an SG tube break. This would allow use of the spray valves in the depressurization process, which could minimize the chance of blowing the pressurizer-relief-tank rupture disk. It should be pointed out that even though the disk failed, there was little discharge to the containment.

4. The industry should consider the problems associated with recovery after a tube break, in particular: (a) Increasing pressure indicates that the level has probably reached its lowest point and is recovering (even if off scale low). (b) Bringing pressure up in the pressurizer to >13.8 MPa (2000 psig) leads to increasing the flow out of the break (and thus to slower recovery) and to decreasing the makeup flow from the SI pumps (due to pump-head-curve characteristics).

5. Evaluate the feasibility of not isolating instrument air to the containment since the power-operated relief valves are used to reduce RCS pressure in this event. Under existing procedures, containment isola-

tion must be reset to repressurize the power-operated relief-valve accumulators.

6. Upgrade procedures for controlling material that goes into and out of the steam generator and other enclosed spaces.

6.2 CRYSTAL RIVER INSTRUMENTATION AND CONTROL FAILURE

[Editor's Note: Information for this article was taken from *Analysis and Evaluation of Crystal River Unit 3 Incident* by the staffs of the Nuclear Safety Analysis Center and the Institute of Nuclear Power Operations.[2] It is clear that "Crystal River was *not* another Three Mile Island."[3] After the incident, refueling was started at the plant, which is owned by Florida Power Corporation, since the core was near the end of its life. Startup was scheduled for June 1980.]

Initial Events

The incident was initiated at ~2:23 p.m. (reference time 0 min) on Feb. 26, 1980, by an instrument and control system electrical malfunction. This malfunction probably was caused by a combination of misaligned connector pins on printed circuit boards and by a technician working on the recently installed saturation-meter circuits which utilized these boards. This failure caused the feedwater control valves to reduce the flow of water to the steam generators significantly. The resulting transient initially increased reactor power somewhat and raised reactor coolant pressure and temperature as one steam generator went dry and the other approached dry-out. At the beginning of the transient, the power-operated relief valve had been opened by the same malfunction in the instrument and control system. Opening the power-operated relief valve did not prevent the reactor coolant pressure from reaching the overpressure-safety-trip point, which initiated the reactor shutdown (time 23 s). The reactor-protection features functioned as designed to shut down the reactor.

Loss of Reactor Coolant

The electrical malfunction caused the power-operated relief valve to stay open. With the reactor shut down and the power-operated relief valve open, the RCS pressure dropped as the reactor coolant discharged into the reactor-coolant drain tank and then into the reactor containment building after the drain tank rupture disk opened. As the RCS pressure dropped, the high-pressure injection pumps came on automatically (at 3 min). The pressure continued to drop. The operator turned off the reactor coolant pumps (at 4 min) and shut the block valve, which stopped the loss of reactor coolant through the power-operated relief valve (at 6–7 min).

The high-pressure injection pumps continued to pump 4200 liters/min (1100 gpm) of water into the RCS. This increased the RCS pressure to the point where a code safety valve opened (10 min), again discharging reactor coolant into the reactor-coolant drain tank and then into the reactor containment building. At 29 min the operator throttled the high-pressure injection system flow to about 950 liters/min (250 gpm), thus reducing the RCS pressure to ~15.9 MPa (2300 psi). Purification system letdown was reestablished (at 30 min), and the makeup-pump recirculating valves were opened (at 33 min). These two operations reduced the amount of water being pumped through the code safety valve into the reactor building by way of the reactor-coolant drain tank; however, the safety valve periodically released coolant at ~15.9 MPa (2300 psi) for ~2 h after the incident began. At 1 h 27 min the high-pressure injection was stopped, and normal makeup and letdown operation was resumed. After 2 h the safety valve stayed seated, and pressure gradually was reduced (at 2 h 15 min) to the 12.8–14.2 MPa (1850–2050 psi) range. The loss-of-coolant portion of the incident was finished in 2 h, after ~150 000 liters (40 000 gal) of coolant had been dumped into the reactor building.

The plant was held in a partially cooled down condition until a pump motor for the low-pressure decay-heat-removal system was repaired. Then the operators followed normal operating procedures to bring the plant to a cold-shutdown condition. The highest pressure reached in the reactor containment building during the incident was ~129 kPa (18.7 psia).

At 5 min, radiation levels of 10 R/h existed in local areas of the reactor containment building. At 15 to 20 min, the radiation levels were up to ~60 R/h. By 30 min the levels were dropping, and at 5 h the radiation levels had decayed to a level of 100–200 mR/h. This radiation could be expected, considering the concentration of radionuclides in the coolant before the incident.

Findings and Conclusions

The following findings, conclusions, and recommendations are excerpted, with minor changes, from the report[2] on the Crystal River incident by the staffs of the Nuclear Safety Analysis Center and the Institute of Nuclear Power Operations.

1. There is no indication of damage to the reactor from the incident at Crystal River on Feb. 26, 1980.

2. The analysis performed by the Nuclear Safety Analysis Center and the Institute of Nuclear Power Operations establishes that the plant was operated in a manner that provided an ample safety margin during the incident (more than adequate to protect the core and to protect the public health and safety).

3. The incident was initiated by an instrument and control-system malfunction. This malfunction probably was caused by an electrical component failure resulting from an undersized plug-in card, which made misalignment of connector pins possible and likely, or by the inadvertent actions of an instrument technician who was working in the area, or by the combined effect of these two circumstances.

4. As noted in item 2 above, the actions of the operators adequately protected the reactor and public health and safety; however, the assessment of system conditions during the incident was made more difficult because about three-quarters of the instruments were aligned to bus NNI-X, which lost power. A more even distribution of instruments between X and Y buses is practicable and would produce greater effective redundance of indications and control.

5. There were no losses or indications of damage to instruments or equipment in the reactor containment building, although a warm, moist environment existed for a time. Relief- and safety-valve discharge of steam into the reactor building built up pressure to about 129 kPa (18.7 psia), and about one-half of the normal radioactivity from the RCS water was transferred to the reactor containment building. The four temperature detectors located in the reactor containment building recorded local temperatures of 60°C (140°F) to 66°C (150°F). Dose rates up to 60 R/h declined to <0.2 R/h in 5 h.

6. Many of the control-room indications went to the mid-scale position upon loss of power. The mid-scale indication occurs by design of the system, which is a −10 to +10 V direct-current control voltage. Since the mid-scale position is usually in the normal operating range, after the power loss, many indications were close to normal values, and thus it was difficult to determine if the readings were true or from a failed (unpowered) instrument. The operators were aware of this and distrusted all readings that were in the mid-scale range, even those from instruments which were functioning properly.

7. The emergency-procedure action levels for declaring class A and B emergencies are rigid. They are possibly too rigid in that they do not allow for any judgment in declaring an emergency. An assessment of the actions during this incident shows:

a. No class A emergency was declared, although it was called for by procedural action levels by 11 min into the incident.

b. The class B emergency should have been declared ~12 min into the incident. There was a delay until ~54 min into the incident.

8. Radioactive releases to the environment as a result of the incident were well below the amounts allowed by government regulations for normal plant operations.

9. The actions, by design, of the integrated control system (ICS) complicated the response of the plant to the partial loss of instrumentation. In response to faulty midrange inputs, the ICS throttled down feedwater flow and caused one steam generator to go dry and the other to approach dry-out, opened the turbine throttle, and pulled control rods until the reactor power level reached the control limit at 103%. These actions were also limited, by design, to tripping the turbine and the reactor. However, this puts the system through unnecessary thermal and hydraulic transients and creates challenges to equipment and operation which should be minimized.

10. The post-Three Mile Island generic emergency procedure for small-break loss-of-coolant accidents (LOCAs) is safe but not necessarily conservative in that it challenges the safety valves and the containment barrier, as in the Crystal River incident. The open power-operated relief valve allowed only a small amount of reactor coolant to enter the containment, whereas the emergency procedure contributed to repeated lifting of the safety valve and caused tens of thousands of liters of primary coolant water to pour into the containment by way of the code safety valve.

11. The steam-generator rupture matrix, which is a safety system to protect against steam and feedwater system ruptures, aggravated this incident by isolating the steam generators when the steam pressure dropped below 4.2 MPa (600 psig). This temporarily denied use of the steam generators as heat sinks even though no rupture existed in the steam-generator system.

Recommendations

Florida Power Corporation should correct items involving training, procedures, plant systems, and hardware discussed below. In a presentation to the Nuclear Regulatory Commission (NRC) on Mar. 4, 1980, Florida Power Corporation described plans for corrective action (immediate, at next refueling, and long term) which respond or relate to all or most of the recommendations made in this report. The timely implementation of such plans is endorsed by this evaluation. Other reactor plant owner–operators should investigate and take corrective actions as required in the following list:

I. Training

I.A Procedural requirements for declaration of appropriate emergencies should be emphasized in plant training sessions.

I.B Review power-supply failures and their effects on control systems. Include such events as ICS-related malfunctions at Crystal River in plant training sessions and in simulator training.

I.C Instrument-technician work practices and their potential impact on plant safety should be reviewed in plant training sessions. Attention should be given to events similar to the transients at the Rancho Seco plant on Mar. 20, 1978, and Jan. 5, 1979, where overcooling resulted from maintenance technician actions.

II. Procedures

II.A Promulgate written procedures for switching instruments between power supplies in the event of power-supply failures, and promulgate a procedure designating the preferred bus for each instrument.

II.B Procedures for steam-generator rupture matrix or its equivalent should be reviewed in conjunction with post-Three Mile Island requirements on steam-driven emergency feedwater pumps to determine if aggravating effects exist during loss of heat sink.

II.C Procedures for orderly plant shutdown following loss of power supply should be prepared or reviewed and revised as necessary. Reactor-system cooldown limits, and the basis for those limits, should be reviewed.

II.D The industry should further analyze and resolve with the NRC the current reactor-coolant-pump trip procedures to be followed during a small-break LOCA. Mandated procedures can be counterproductive to safety if they are not sufficiently discriminating to specific circumstances and specific plant designs.

II.E The industry should review the current high-pressure injection pump requirements and resolve any procedural issues with the NRC. Procedures that avoid or minimize challenges to safety valves, primary systems, and eventually to the containment building itself are needed. Mandated procedures can be counterproductive to safety if they are not sufficiently discriminating to specific circumstances and plant designs.

II.F Procedures for the declaration of emergencies should be reviewed to determine if responsibility for monitoring plant conditions that lead to the declaration of a specific emergency category should be assigned to a specific individual. It is suggested that this individual would also be responsible for immediately informing the senior person in charge at the time when the conditions for emergencies and emergency notification have been met.

III. Plant Systems and Hardware

The following list of problems should be investigated and corrective action taken as required.

III.A Loss of Power Supply

III.A.1 Need for backup or bus transfer capabilities if a fault trips the instrumentation and control-power supplies.

III.A.2 Coupling of indication, control, and computer input signals, e.g., loss of power to the integrated control system, nonnuclear instrumentation, and the reactor cooling system results in a loss of control-board indication of many signals.

III.A.3 Opening of the power-operated relief valve and its failure modes due to voltage variation, which results in loss of proper set-point reference.

III.A.4 Susceptibility of control systems to incorrect information caused by electrical faults, e.g., choking off feedwater to steam generators, withdrawing rods, and opening the turbine throttle.

III.A.5 Instrument loops are selected by a switch in the control room. Designs should be reviewed (and, wherever practical, field tested) to determine the effects of a loss of power to one of the instrument loops and to establish the absence of cross-contamination of multiple power supplies in the instrument and control functions.

III.A.6 The coincidence of having a mid-scale operating point and mid-scale instrument failure on loss of power gives uncertain information, e.g., the loss of emergency feedwater automatic start because the steam-generator level indication appeared to be higher than actual.

III.A.7 Assignment of instruments to specific buses should ensure as much redundancy as possible.

III.B Data Handling and Display

III.B.1 The adequacy of data handling and display systems should be reviewed. Examples of specific problems encountered during the incident at Crystal River on Feb. 26, 1980, were: (*a*) Many instances of alarm conditions returning to a normal state without any prior indication of having reached an alarm state. (*b*) Loss of computer printout due to overload. (*c*) The system monitoring the in-core temperatures automatically prints any temperature in excess of 4.8 MPa (700°F). The basis for selecting 4.8 MPa (700°F) should be reviewed to determine if this number should be revised since data were lost during the transient.

III.B.2 Plant transient monitoring and recording. Plant transient records which are independent of process computer and which provide a tape record of main plant parameters are desirable for all plants. They are desirable on an earliest practicable schedule.

III.C Steam-Generator System

III.C.1 The steam-generator rupture matrix or equivalent should be reviewed and changed as necessary to prevent actuation of isolation and loss of heat sink for events that do not actually involve ruptures in the steam-generator system.

REFERENCES

1. Northern States Power Company, Prairie Island Nuclear Generating Plant, *Prairie Island Unit 1 Steam Generator Tube Break—Licensee Event Report Update,* Report RO-79-027-1, Feb. 19, 1980.
2. Nuclear Safety Analysis Center and the Institute of Nuclear Power Operations, *Analysis and Evaluation of Crystal River Unit 3 Incident,* Report NSAC-3 (INPO-1), NTIS, March 1980.
3. Crystal River: "Not" Another Three Mile Island, *Lab News,* Oak Ridge National Laboratory, March 1980.