TERA
50-320

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

MAY 15 1980

Mr. James R. Morrison
12432 Cantebury
Warren, Michigan 48093

Dear Mr. Morrison:

This is in reply to your letter of May 16, 1979, regarding nuclear accidents before Three Mile Island. I am sorry for the long delay in responding, but we have been very busy with the aftermath of the Three Mile Island accident.

Enclosed for your information are the following documents:

> Report NUREG-0572 of September 1979 on "Review of Licensee Event Reports (1976-1978)" by the Advisory Committee on Reactor Safeguards of the Nuclear Regulatory Commission.
>
> A section on "Past Accidents in Nuclear Reactor Facilities and Nuclear Power Plant Airborne Radioactivity Releases" excerpted from the Summary of the Technical Staff Analysis Report to the President's Commission on the Accident at Three Mile Island of October 1979.
>
> Excerpts from a section on "Precursor Events" from Volume II, Part I, of "Three Mile Island -- A Report to the Commissioners and to the Public" by a Special Inquiry Group of the Nuclear Regulatory Commission, January 1980. (Four specific incidents prior to the Three Mile Island accident are discussed.)

I trust that this material will be of interest to you.

Sincerely,

Harold Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosures:
As stated

8007210046 H

TECHNICAL STAFF ANALYSIS REPORT

SUMMARY

TO

PRESIDENT'S COMMISSION ON THE
ACCIDENT AT THREE MILE ISLAND

ADVANCE COPY

NOT FOR PUBLIC RELEASE

BEFORE AMs, WEDNESDAY, OCTOBER 31, 1979

EXCERPT

PAST ACCIDENTS IN NUCLEAR REACTOR FACILITIES AND NUCLEAR POWER PLANT AIRBORNE RADIOACTIVITY RELEASES

Part I of this report describes selected accidents which have occurred in nuclear reactor facilities worldwide. Included are accidents involving central station power plants, plutonium production reactors, demonstration plants, and experimental and research reactors. The condition for inclusion in this compilation is that the accident fulfill one of the following criteria:

- o Caused death or significant injury
- o Released significant radioactivity offsite
- o Results in core damage
- o Causes severe damage to major equipment
- o Was a precursor to a potentially serious accident
- o Resulted in inadvertent criticality
- o Resulted in significant recovery cost

Of the 40 accidents considered, 22 resulted from equipment failure,10 from human failure, and 7 involved both equipment failure and human failure.

By type, there were 27 nuclear accidents and 13 non-nuclear. The latter are defined as cases where criticality of the core was not a factor; either the reactor was unfueled, shutdown, or systems not associated directly with reactor operation were involved. In this type of accident two people were killed and 8 injured. Radioactive release accompanied three of these but in no case was it significant offsite. All of the nonnuclear accidents involved central station power plants.

Nuclear type accidents to central station power plants resulted in no personnel injuries or deaths. Three Mile Island received by far the most attention because of the nature and duration of the accident and the number of people involved. It was the first central station power plant accident to release more than trivial amounts of radiation. Inadvertent criticality at two power plants did not release any activity or cause any core damage.

The most serious accident radiologically happened to the Windscale productionreactor in England when art of the uranium-graphite core was destroyed by a smoldering fire. Milk consumption in a 200 square mile area was restricted because of iodine contamination through animal feed.

The only serious criticality accidents have occurred with experimental and research reactors. In one of these where were three fatalities and the reactor was destroyed. Serious core damage was incurred at four other reactors of this type when they became supercritical. No serious off-site contamination resulted, however, for any of these accidents.

In the second part of this report, yearly releases of noble gases and halogens are tabulated for power plants operating in the United States. Some of the higher routine yearly releases from operating nuclear power stations have been comparable to the single event release of the Three Mile Island accident.

II. Nuclear Power Plant Airborne Radioactivity Releases

The release of fission products from the TMI-2 accident consisted of 2.5 million curies of noble gases, primarily xenon and about 15 curies of iodine-131. The question may be asked, how does this short-term, single-event release compare with the historical record of allowed annual release of fission products from operating reactors? The following two tables present information taken from NRC reports1,2/, concerning routine releases of noble gases, halogens and particulates from operating nuclear reactors in the United States Annual releases that are comparable to releases resulting from the TMI-2 accident are underlined.

We point out that in 1975 and 1976 amendments to 10 CFR Part 50 (Appendix I) severely limited the allowed releases from routine operations. The concept of "as low as practicable" releases required power stations to install equipment limiting releases to low values.

The release of radioactive noble gases from TMI-2 led to a low average radiation dose to individuals in the neighborhood and to a collective dose to the total population within a 50-mile radius of about 3,300 person-rem.3,5/ For comparison purposes the population doses from operating nuclear power plants in 1975 has been estimated.4/ These ranged from a high of 750 person-rem to a low of 0.008 person-rem.

1/ NUREG-0077, "Radioactive Materials Released from Nuclear Power Plants, 1974, U.S. NRC, June 1976.

2/ NUREG-0521, "Radioactive Materials Released from Nuclear Power Plants, Annual Report 1977," U.S. NRC, Jan. 1979.

3/ Report of the TMI Ad Hoc Population Dose Assessment Group, "Population Dose and Health Impact the Accident on the Three Mile Island Nuclear Station", May 10, 1979.

4/ "Population Dose Commitment Due to Radioactive Releases from Nuclear Power Plant Sites in 1975", PNL-2439, October, 1977, by Baker, Soldat and Watson.

5/ "President's Commission on the Accident at Three Mile Island - Report of the Task Group on Health Physics and Dosimetry", J.A. Auxier, et al. Sept. 28, 1979 gives an estimated collective dose of 2,800 person-rem.

NUCLEAR POWER PLANT AIRBORNE RELEASES[1, 2]

Curies of Noble Gases (Kr, Xe, etc.)

Plant	1970	1971	1972	1973	1974	1975	1976	1977
Big Rock Point 1	280,000	284,000	258,000	230,000	188,000	50,600	15,200	13,400
Browns Ferry 1, 2, 3	--	--	--	--	64,000	92,400	<80,500	<166,000
Cooper Station	--	--	--	--	2,000	19,800	38,000	1,270
Dresden 1	900,000	753,000	877,000	840,000	98,000	520,000	452,000	520,000
Dresden 2, 3	--	580,000	429,000	880,000	627,000	369,000	323,000	313,000
Humboldt Bay 3	540,000	514,000	430,000	[illegible]50,000	572,000	297,000	93,000	?
Lacrosse	1,000	1,000	31,000	91,000	49,000	57,100	124,000	42,500
Millstone Point 1	--	276,000	726,000	79,000	912,000	2,970,000	507,000	620,000
Monticello	--	76,000	751,000	870,000	1,480,000	155,000	11,400	6,870
Nine Mile Point 1	10,000	253,000	571,000	872,000	558,000	1,300,000	176,000	3,530
Oyster Creek	110,000	516,000	866,000	810,000	279,000	206,000	167,000	177,000
Peach Bottom 2, 3	--	--	--	<1,000	<1,000	13,000	209,000	71,100
Pilgrim 1	--	--	18,000	230,000	546,000	46,000	183,000	413,000
Quad Cities 1, 2	--	--	132,000	900,000	950,000	110,000	33,600	25,600
Vermont Yankee	--	--	55,000	180,000	64,000	4,080	3,030	3,350
Arkansas 1	--	--	--	--	196	1,030	5,690	13,900

Connecticut Yankee	1	3	1	32	7	480	452	3,120
Fort Calhoun	--	--	--	67	303	429	1,940	3,810
H. B. Robinson	--	<1	<1	3,100	2,310	1,170	640	476
Indian Point 1	2	<1	1	122	611	--	--	--
Indian Point 2	--	--	--	15	5,580	8,200	11,600	16,000
Kewaunee	--	--	--	--	3,350	2,450	1,400	2,430
Maine Yankee	--	--	<1	161	6,360	4,090	1,300	286
Oconee 1, 2, 3	--	--	--	9,300	19,400	15,100	43,900	35,600
Palisades	--	--	1	454	<1	2,610	30	60
Point Beach 1, 2	--	1	3	5,750	9,740	44,500	1,910	1,130
Prairie Island 1, 2	--	--	--	9	358	2,170	1,740	673
R. E. Ginna	10	32	12	576	757	10,400	5,520	3,200
San Onofre 1	<1	8	19	11,000	1,780	1,110	416	154
Surry 1, 2	--	--	<1	866	55,000	8,040	19,100	19,000
Three Mile Island 1	--	--	--	--	916	3,630	2,760	16,600
Turkey Point 3, 4	--	--	--	530	4,660	13,400	15,600	23,300
Yankee Rowe	<1	<1	<1	35	40	22	26	12
Zion 1, 2	--	--	--	4	2,990	48,000	114,000	2,200

21-5

NUCLEAR POWER PLANT AIRBORNE RELEASES[1, 2]

Curies of Halogens and Particulates (half life ≥ 8 days)

Plant	1970	1971	1972	1973	1974	1975	1976	1977
Big Rock Point 1	0.13	0.61	0.15	4.60	0.16	0.12	0.05	0.01
Browns Ferry 1, 2, 3	--	--	--	--	0.12	0.27	< 0.07	0.10
Cooper Station	--	--	--	--	0.24	0.05	< 0.04	< 0.02
Dresden 1	3.3	0.67	2.75	0.04	0.68	0.96	0.84	4.93
Dresden 2, 3	1.6	8.68	5.89	6.70	6.60	4.31	5.49	6.86
Humboldt Bay	0.35	0.3	0.48	0.29	0.84	1.06	0.08	0.004
Lacrosse	< 0.06	< 0.01	0.71	0.20	0.04	0.10	< 0.07	0.17
Millstone Point 1	--	4.0	1.32	0.20	3.26	9.98	2.33	4.86
Monticello	--	0.05	0.59	1.20	5.69	3.71	0.17	0.08
Nine Mile Point 1	< 0.01	0.06	0.97	1.98	0.89	2.78	2.20	0.20
Oyster Creek	0.32	2.14	6.48	7.02	3.51	5.64	6.39	9.05
Peach Bottom 2, 3	--	--	--	< 0.01	0.01	0.04	0.98	0.27
Pilgrim 1	--	--	0.03	0.47	1.45	2.58	0.67	0.69
Quad Cities 1, 2	--	--	0.75	5.5	8.88	1.31	1.33	1.69
Vermont Yankee	--	--	0.17	0.07	0.36	0.01	< 0.01	0.01
Arkansas 1	--	--	--	--	0.05	0.74	0.06	0.01

Connecticut Yankee	<0.01	0.03	0.02	0.05	<0.01	<0.01	<0.01	0.002
Fort Calhoun	--	--	--	<0.01	<0.01	<0.01	<0.02	0.01
H. B. Robinson	--	0	0.03	0.30	0.05	0.02	0.10	0.004
Indian Point 1	0.08	0.21	0.93	0.01	0.11	--	--	--
Indian Point 2	--	--	--	<.01	0.43	1.62	0.24	0.06
Kewaunee	--	--	--	--	0.02	0.66	< 0.01	0.02
Maine Yankee	--	--	<0.01	0.94	0.05	<0.01	<0.01	0.005
Oconee 1, 2, 3	--	--	--	0.01	0.03	0.01	0.27	0.54
Palisades	--	--	<0.01	0.31	0.01	0.30	0.04	0.01
Point Beach 1, 2	--	<0.01	0.03	0.55	0.16	0.07	0.02	0.005
Prairie Island 1, 2	--	--	--	<0.01	<0.01	0.02	0.01	0.008
R. E. Ginna	0.05	0.17	0.04	<0.01	<0.01	0.02	0.03	0.03
San Onofre 1	<0.01	<0.01	<0.01	1.61	<0.01	0.04	<0.01	0.0002
Surry 1, 2	--	--	<0.01	0.04	0.14	0.05	0.35	0.12
Three Mile Island 1	--	--	--	--	<0.01	<0.01	0.01	0.03
Turkey Point 3, 4	--	--	--	0.06	3.63	0.43	0.42	1.04
Yankee Rowe	<0.01	<0.01	<0.01	0.19	0.63	0.01	<0.01	0.0001
Zion 1, 2	--	--	--	<0.01	0.01	0.14	0.09	0.05

21-7

21-7

DRAFT

VOLUME II, PART 1

THREE MILE ISLAND

NOT FOR RELEASE
BEFORE 1:30 P.M. E.S.T.
JANUARY 24, 1980

A Report to the
Commissioners
and to the Public

DRAFT

Mitchell Rogovin
Director

George T. Frampton, Jr.
Deputy Director

Nuclear Regulatory Commission
Special Inquiry Group

DRAFT

EXCERPTS

22. Miller dep. at 55.

23. Kohler dep. at 75.

24. Walters dep. at 29.

25. Hallman dep. at 50.

26. Hallman dep. at 50; Dunn dep. at 69; Walters dep. at 31.

27. Hallman dep. at 52.

28. Murray dep. at 64.

29. Ibid at 66.

30. Walters dep. at 32-33.

31. Taylor dep. at 61.

32. Miller dep. at 55.

33. Murray dep. at 64.

I.

C. PRECURSOR EVENTS

OVERVIEW AND GENERAL DESCRIPTION

The experience of the nuclear power industry and the NRC with accidents and episodes presaging the TMI-2 accident was of particular interest to the Special Inquiry Group. Several such events occurred during the preceding 8 years in connection with plants other than the TMI installations. One problem at TMI-2 was also a possible precursor to the March 28, 1979 accident.

The history of the industry was reviewed to determine (1) if it contained useful foreknowledge of the March 28, 1979 problems at TMI, (2) whether the information was effectively evaluated and disseminated, and (3) whether that information was ultimately effectively utilized.

Initially, the Special Inquiry Group planned to investigate all potential precursor events to determine their relevance and significance and how they were handled. However, as work progressed we realized that there were a number of additional events and issues that although they did not appear to be significant, might have yielded information that would substantiate the observations we made as a result of our review of the precursors that we did investigate. These events were not addressed because the resources required to investigate these peripheral issues were not justified by the expected return. Therefore, the precursors discussed in this report are best described as a representative sample of all the precursor events associated with the accident at TMI-2. We believe that this sample accurately reflects the ways that these events and issues have been handled.

The more significant precursor matters examined begin with a 1971 letter to the Atomic Energy Commission from H. Dopchie of Belgium (see Section IC4) which noted a problem with pressurizer level after a small-break loss-of-coolant accident from the pressurizer steam space of a Westinghouse pressurized water reactor. In 1974, such an event occurred at a Westinghouse reactor (NOK-1) at Beznau, Switzerland (see Section IC5).

In 1975, the Nuclear Regulatory Commission published a report of a detailed 3-year study, variously known as "WASH-1400", "The Reactor Safety Study" or "The Rasmussen Report", which attempted to measure the risks in the operation of nuclear reactors; small-break loss-of-coolant accidents and small releases of radioactivity were included (see Section IC6).

In September of 1977 the Davis Besse nuclear powerplant of the Toledo Edison Company, designed by Babcock and Wilcox (B&W), had a transient that was very similar to the TMI-2 accident (see Section IC8).

At the same time, at the Tennessee Valley Authority (TVA), Carl Michelson, a nuclear engineer and a consultant to the NRC's Advisory Committee on Reactor Safeguards (ACRS), raised to his TVA superiors some long-considered concerns about the susceptibility of Babcock and Wilcox designed plants to very-small-break loss-of-coolant accidents (see Section IC7). TVA submitted the Michelson report to Babcock and Wilcox for analysis in April of 1978. A handwritten copy had been given informally in the fall of 1977 by Michelson to Jesse Ebersole, a close personal friend and a member of the ACRS. Ebersole, in the process of preparing questions that were eventually sent to Portland General Electric Company about its Pebble Springs, Oregon, plant used Michelson's report as the basis for a question about operator interpretation of pressurizer level in a B&W plant during a loss-of-coolant accident (see Section IC10).

At Babcock and Wilcox Company Nuclear Power Generation Division headquarters, a concern arising out of the incident in September 1977 at the Davis Besse plant prompted engineer Joseph J. Kelly (in the Plan Integration Section) on November 1, 1977, and Bert M. Dunn, (Chief of the Emergency Core Cooling Systems Analysis Branch) on February 9, 1978 to urge their management to revise guidance concerning operator instructions on stopping the High Pressure Injection pumps during accidents (see Section IC9).

At the NRC, perhaps as a outgrowth of the composite impact of the September 1977 Davis Besse incident, the Michelson report, and Ebersole's Pebble Springs questions, Sanford Israel of the Reactor Systems Branch of the Office of Nuclear Reactor Regulation prepared a note signed on January 10, 1978, by his Branch Chief, Thomas M. Novak, concerning pressurizer design in B&W plants. The note urged that reviewers verify that operators of future plants be provided adequate information about procedures for terminating High Pressure Injection flow (see Section IC12).

In March 1978, D. M. Sternberg in Region I, Office of Inspection and Enforcement (I&E), reported to K. V. Seyfrit in I&E Headquarters that TMI-2 had experienced a blowdown (after a reactor trip) on March 29, 1978 because a pressurizer Pilot Operated Relief Valve (PORV) opened after a loss of control power (see Section IC14).

An event on March 20, 1978 at the Rancho Seco nuclear power plant near Sacramento, California, involving loss of power to some nonnuclear instrumentation, prompted concerns at B&W about the necessity for operator education on procedures to follow when such loss of instrumentation occurs. B&W wrote to all its Site Operations Managers (except TMI) that "pressurizer level and RCS pressure assure that the Reactor Coolant System is filled..." 1 (emphasis added) (see Section IC13).

At NRC's Region III, James C. Creswell, Reactor Inspector, who was an inspector for Davis Besse, developed a series of concerns, six of which he submitted on January 8, 1979, through channels for review by the Atomic Safety and Licensing Board, and some of which he personally laid before Commissioners Bradford and Ahearne in March of 1979 (see Section IC11). Figure IC-1 is a graphical representation of the significant precursor milestones. Figure IC-2 is a graphical representation of the organizational relationship of NRC employees who were directly involved with precursor events or issues.

This chapter reviews these events in detail and gives the Special Inquiry Group's conclusions and recommendations.

I. CONCLUSIONS AND RECOMMENDATIONS

(a) The nuclear industry and the NRC had little or no concern about what the operators saw during a transient and what they did as a result. Actual plant operating and emergency procedures were not reviewed in any systematic fashion by the NRC or by the vendor. Incidents were assessed almost entirely from the perspective of the hardware with little concern about what the operator saw or did.

In the design of equipment, much consideration is given to why a piece of equipment will not perform an anticipated function, (e.g., why a valve will not open when it should). However, little consideration need be given to why a piece of equipment might perform a function when passivity is expected. For equipment, this emphasis is proper because a piece of equipment is more likely to fail to perform a required function, than to activate and perform a function for no apparent reason. This logic has been erroneously applied to the operator. However, people by nature are not passive. The operators have shown a strong willingness to become actively involved in operating the plant following an incident. Once the operators decide that they are going to take an active role in a particular event, they have shown themselves to be very persistent and innovative in finding a way to get a certain function done. However, defining all of the reasons why an operator might initiate an action has received much less attention than it should have received during the design and licensing of nuclear power plants. Therefore, with machines, the concern is that the machines will not perform when they should; but with operators, the concern should be that the operator will perform when they should not.

In the past, the operators have been essentially ignored by the NRC and by the plant designers. On the other hand, incidents such as the one that occurred at Davis Besse on September 24, 1977 make it quite clear that operators do not consider themselves to be passive observers during an incident. The operators are an active component. Moreover, they can and do intervene in the automatic features of the plant as well. Such intervention may be right or very wrong.

If it is decided that the operators should play an active role in mitigating and minimizing the consequences of an

5. Beznau* Incident--August 20, 1974

On August 20, 1974, an incident occurred at the NOK-1* Nuclear Power Plant in Beznau, Switzerland that bears some similarity to the accident that subsequently occurred at Three Mile Island. The NOK-1 plant was designed by Westinghouse. The design is similar to nuclear power plants that were built by Westinghouse in the United States.

The particular incident in question began with the reactor operating at 100 percent power. A trip of one of the two turbine generators occurred. As a result, the Reactor Coolant Sytem termperature and pressure increased rapidly and both PORVs opened. One PORV failed to close and a subsequent depressurization of the Reactor Coolant System occurred. The reactor tripped on low pressure as a result of this depressurization. As pressure continued to decrease, steam formed in the Reactor Coolant System hot leg and pressurizer level began to rise. It eventually increased past the 100 percent point and remained off-scale for 3-5 minutes. The operators were able to identify that the PORV was open in approximately 2-3 minutes and shut the isolation valve (there is no indication of what caused the operators to realize in such a short period of time that the PORV was open). After the PORV was shut, the pressurizer level fell rapidly as the steam bubbles in the Reactor Coolant System collapsed. Finally, approximately 12 minutes into the incident, the pressurizer level reached the five percent point and High Pressure Injection was initiated.

In this particular design, a coincident initiation was required for High Pressure Injection actuation. This initiation required both a low Reactor Coolant System pressure and a low pressurizer level. Therefore, because the pressurizer level went off-scale high due to void formation in the Reactor Coolant System, the pressurizer level did not decrease initially and did not cause High Pressure Injection to begin until 12 minutes into the incident.

* This incident has come to be known as the Beznau incident. In fact, the reactor, which is located in Beznau, Switzerland, is named "NOK-1". There is no "Beznau Reactor".

The incident was analyzed by a team from Westinghouse's Brussel, Belgium office and a report prepared. This report was distributed to various individuals in the Westinghouse domestic reactor offices in Pittsburgh, Pa. The analysis indicated that all existing protection systems had performed properly. [12]

This conclusion was based in part on an analysis of a small LOCA from the steam space in the pressurizer which had been performed in 1971. This analysis showed that during such an event, pressurizer level would rise and prevent automatic initiation of High Pressure Injection. [13] The analysis also showed that the operators had approximately 50 minutes to manually initiate High Pressure Injection before core damage would begin. [14] Westinghouse concluded that this amount of time (20 minutes is normally considered an adequate period for an operator to take required manual actions) and the indication available to the operator (Westinhouse plants have, among other indications, direct indication of the PORV position) were sufficient to provided adequate protection. 15 This conclusion was substantiated by the fact that the operators at Beznau isolated the PORV in 2 to 3 minutes.

It should be noted that prior to the TMI accident, Westinghouse guidance to utilities concerning small LOCA procedures did not provide specific warnings that pressurizer level might increase during such an event. The Westinghouse operator training program included a stuck-open PORV and the operator was instructed how to recognize this event. However, the Westinghouse simulator did not indicate a rising pressurizer level, but only indicated a more slowly decreasing level. 16

The results of the 1971 anaylsis had been documented to the AEC in the Safety Analysis Report (Amendment 1 dated October 1972) for the RESAR-3 standard plant. 17 Although this report did not specifically state that the pressurizer level would increase during such an event, it did state that for breaks in the 2 to 6 inch range, High Pressure Injection might not result. The report also noted that a delay of High Pressure Injection of more than 50 minutes would not result in core uncovering.

Beginning with RESAR-3 the standard Westinghouse design, was changed to require only low pressure to initiate High Pressure Injection. This change was primarily the result of operating experience which indicated that spurious actuation of High Pressure Injection would not be a problem if the coincident pressure and level requirement was eliminated. Westinghouse considered changing older designs, but decided that because of the time and indication available to the operator, backfitting of this change was not required. 18

The original report of the Beznau incident was not submitted to the AEC at the time that it was prepared because the plant had responded as expected. The NRC eventually became aware of the incident at Beznau during discussions with Westinghouse employees following the TMI accident. The NRC subsequently obtained from the Swiss government the Westinghouse report and another report prepared by the Swiss. Paradoxically, however, because of the current regulatory requirements with respect to Proprietary Information, the Swiss government was able to designate this information as Proprietary which would have prevented the dissemination of the details of this event to the public. In fact, it was initially intended that the only reference that would be made in any public NRC documents with respect to the Beznau incident, was a statement that had been approved by the Swiss government. This statement said, "We are aware of one incident at a foreign reactor designed by Westinghouse which occurred a number of years ago in which a PORV was challenged during a turbine trip transient and failed to reclose when pressure decreased. The failure to close was detected in a few minutes by the operators who immediately isolated the valve by closing the blocked valve in series with the PORV. This action terminated the incident. The failure to reclose was due to the rupture of the cast iron frame between the valve operator and the valve body which was caused by a water slug hitting the valve. The source of the water slug was the loop seal located between the pressurizer and the relief valve. Investigation of this event identified the cause of the valve failure to be design error which, we understand, has been subsequently remedied." 19 There is no indication in this statement that pressurizer level failed to decrease and that High Pressure Injection was inhibited as a result of the response of the plant. It was only after the inappropriateness of the withholding of this information from the public, was raised by a number of individuals, including members of this Special Inquiry, that the proprietary restrictions were removed.

After the accident at TMI, Westinghouse provided guidance to plants that still have the coincident low pressurizer/low pressurizer level High Pressure Injection. This guidance pointed out that during small LOCAs from the pressurizer, there may be a problem with pressurizer level hanging up. By letter date April 10, 1979 (Ref. 20), Westinghouse informed the NRC that they had advised utilities that the problem could exist and they were recommending that the operators be specifically instructed to monitor pressure and manually initiate High Pressure Injection if pressure dropped below the actuation point.

Specific Conclusions

(a) An incident occurred at the NOK-1 nuclear plant in 1974 that demonstrated the phenomenon of increasing pressurizer level during a small loss-of-coolant accident from the steam space in the pressurizer. This phenomenon was subsequently observed at the Davis Besse plant in September 1977, and during the TMI accident. In the specific case of the Beznau incident, the high pressurizer level caused the High Pressure Injection to fail to initiate. At Davis Besse and TMI, the High Pressure Injection system initiated but was subsequently stopped because the operators erroneously interpreted the high pressurizer level.

(b) The relevant phenomenon (i.e., increasing pressurizer level during a small LOCA from the pressurizer steam space) had been previously identified by Westinghouse. Therefore, the plant responded as expected. The implications of this phenomenon but not the phenomenon itself, had been reported to the AEC prior to the Beznau incident. It is not known how clearly the AEC recognized this phenomenon as a result of this matter. However, it does appear that the AEC was never explicitly informed that for older Westinghouse designs (i.e., prior to RESAR-3) operator action was required during a small LOCA from the steam space in the pressurizer. As a result, it was not possible for the AEC to incorporate the lessons that might have been learned from this incident into the licensing of Westinghouse plants or PWRs in general.

(c) Because of the restrictive nature of the current regulations with respect to proprietary information received from foreign governments, it is very possible that the information contained in the Beznau report would not have become part of the public record even in light of the TMI accident. However, it must be recognized that there is a trade-off between restrictive proprietary information provisions that allow a foreign government to provide information that will subsequently not become part of the public record; and the fact that if foreign governments can no longer provide this information with confidence that it will not become public, they will refuse to provide the information in the future.

6. Reactor Safety Study (WASH-1400)--October 1975

In 1975 the NRC published the results of an extensive three year study which attempted to quantify the risks associated with operation of a nuclear reactor. The report was formally titled, "The Reactor Safety Study." 21 It has also come to be known as "WASH-1400" or "the Rasmussen Report".

WASH-1400 is a precursor to the accident at TMI for a number of reasons.

First, WASH-1400 identified the category of small-break LOCAs as one of the most significant contributors to the risk from nuclear reactor operation. 22,23 Of particular concern were the smallest class of Reactor Coolant System breaks (1/2 inch to 2 inches effective diameter) which included a break equivalent to the stuck open PORV at TMI (approximately 1 inch effective diameter). This dominance of very small LOCAs over larger LOCAs was found even in the most serious (with respect to radioactivity releases from the containment) categories of accidents identified in WASH-1400. For example, the probability of the most serious category of accident assessed in WASH-1400 being initiated by a very small LOCA is 50 times greater than the probability that it would be initiated by a large LOCA. 24 This dominance was due primarily to the fact that small pipes are considerably more common than large pipes, and large pipes are installed using stricter codes and requirements. 25

Despite this emphasis in WASH-1400 on the significance of small LOCAs, the NRC continued to place a great deal of emphasis in the licensing process and in research allocations, on large LOCAs. 26 Had the emphasis been shifted to these very small LOCAs, it is possible that a better understanding of the subsequent events at Davis Besse (September 24, 1977) and at TMI might have been developed.

Second, WASH-1400 emphasized that small releases of radioactivity resulting from various plant accidents are much more likely than large catastrophic failures releasing large quantities of radioactivity. For example, the least severe category of accident consequences (Category 9), which includes the level of releases that occurred at TMI, 27 was found to be over 400 times more likely than the most severe category (e.g., Category 1). 28

As a result of this conclusion, the NRC should have recognized that these less severe accidents deserved a significant emphasis in the regulatory process because the probabilities indicated that an event of this type would occur in the coming years. As has been shown by the accident at TMI, increased emphasis should have been placed on emergency planning and dissemination of information during such high probability but low consequence events. This is particularly true when one recognizes that although the radioactivity released during these events did not produce a significant physical health effect, the psychological stress caused by draft.

(d) Michelson and Ebersole were painfully naive to believe that a handwritten draft report, informally handed to a first-line supervisor within the NRC, would receive anything more than a cursory review. The lack of followup by Ebersole after he forwarded the report to Israel exacerbated the problem of this report not being given extensive consideration by the NRC.

(e) B&W response to the Michelson report was excessively slow. However, this slow response was due primarily to the fact that B&W believed that the technical issues raised in the report were not significant and were already adequately addressed in earlier analyses, and that the bulk of their effort was associated with explaining why the concerns raised in the report were not significant issues.

(f) With respect to the issue of operator interpretation of pressurizer level, B&W felt that this issue had been resolved by virtue of the additional guidance that Kelly, Jones, and Dunn all mistakenly believed had been sent to the various utilities as a result of the Kelly-Dunn memos.

(g) Although Michelson was (and still is) a consultant to the ACRS, he did not provide the Michelson report to Ebersole, a member of the ACRS, because of this formal relationship (i.e., the report was not submitted to the ACRS). Michelson and Ebersole had been close personal friends since long before either of them became associated with the ACRS. It was in this context of personal friends who shared a common interest (i.e., small-break LOCAs) that the Michelson report was given to Ebersole.

8. Davis Besse--September 24, 1977

An incident occurred at the Davis Besse Nuclear Power Station* on September 24, 1977, that bears a strong resemblance to the subsequent accident at TMI-2. The incident began at 9:34 p.m. while the plant was operating at 9% power with one Effective Full Power Day of operation. The incident was initiated by a spurious half-trip of the Steam and Feedwater Rupture Control System. This trip stopped the feedwater flow to the No. 2 steam generator which caused the level in the steam generator to decrease. At 1 minute and 16 seconds after the spurious half-trip, a full trip was initiated as a result of low level in the No. 2 steam generator. This full trip isolated the main feedwater flow to the other steam generator and initiated auxiliary feedwater flow. However, the No. 2 auxiliary feedwater pump turbine did not come up to full speed because of binding of the turbine governor. This situation resulted in no auxiliary feedwater flow to the No. 2 steam generator. At approximately the same time that the full trip of the Steam and Feedwater Rupture Control System occurred, the Pilot Operated Relief Valve (PORV) opened as designed. However, due to a missing relay in the control circuit, the valve rapidly cycled open and shut, and eventually failed in the open position.

* On August 1, 1969, the Toledo Edison Company and the Cleveland Electric Illuminating Company, two privately

owned public utility companies, applied for a license to construct and operate the Davis Besse Nuclear Power Station, Unit 1 located on Lake Erie about 21 miles east of Toledo, Ohio. A Construction Permit was issued on March 24, 1971, and by letter dated April 21, 1977, the Nuclear Regulatory Commission issued an Operating License. By September 24, 1977, the plant was still in the startup testing program.

The design of the Nuclear Steam Supply System by Babcock & Wilcox is similar to TMI-2. Engineered safety features built into the Davis Besse Unit 1 included an emergency core cooling system with a core flood system and both high and low pressure injection systems. The architect-engineer for Davis Besse was Bechtel Corporation. The turbine generator was supplied by General Electric Company.

The full trip of the Steam and Feedwater Rupture Control System also shut the Main Steam Isolation Valves. As a result of the loss of cooling to the Reactor Coolant System, the Reactor Coolant System temperature increased, which in turn caused pressurizer level to increase sharply. At 1 minute and 47 seconds the operator manually tripped the reactor because of high pressurizer level.

The tripping of the reactor, the open PORV, and the injection of cold auxiliary feedwater to the No. 1 steam generator caused Reactor Coolant System temperature and pressurizer level to decrease. At this point, the operators were verifying proper operation of various safety features and responding to numerous alarms that were received in the control room. The alarms were received so rapidly that the implications of each alarm could not be analyzed in detail. The difficulties were further compounded by the fact that the operators did not immediately realize that the incident had been initiated by a malfunction of the Steam and Feedwater Rupture Control System. [80]

As pressure continued to decrease, it eventually reached 1600 psi (at approximately 3 minutes), at which point the Safety Features Actuation System actuated. The actuation caused containment isolation and initiated High Pressure Injection flow. The containment isolation shut the vent on the quench tank, which received the discharge from the open PORV. As a result, the pressure increased in the quench tank and caused the rupture disk to blow. The operators realized that the rupture disk had blown. However, they thought that, at most, the PORV had stayed open slightly longer than normal; they did not realize that the PORV was still stuck open.

The operators did have the computer printout of temperature at the outlet of the PORV available; however, they did not use that information because the alarm printer was too far behind. [81] The only other indication of the PORV position was from the control power signal for the solenoid, and that erroneously indicated that the valve was shut.

At approximately 4-1/2 minutes pressurizer level stopped decreasing and began to increase as a result of the influence of the High Pressure Injection pumps. However, Reactor Coolant System temperature and pressure continued to decrease. At approximately 6 minutes, the operators stopped the High Pressure Injection pumps because pressurizer level had returned to normal and, in fact, had increased above the initial level. 82 Securing the High Pressure Injection was consistent with the plant's emergency procedures, which stated in Emergency Procedure 1201.06.2, Section 2.4.3, "Note that as RCS |reactor coolant system¬ pressure is decreased, the HPI |high pressure injection¬ must be throttled to maintain pressurizer level." 83 However, the action of stopping High Pressure Injection was inconsistent with the plant operating procedures, specifically Plant Procedure 1101.01.2, Section 1.1.3, Item 6, which states,

> "Reactor coolant system pressure must be maintained above the pressure that would allow the formation of a steam bubble at the highest point of the 36-inch reactor coolant piping." 84

In hindsight, some of the operators were amazed that they stopped High Pressure Injection based on pressurizer level indication alone, because they realized that the plant was approaching saturation conditions. They can only attribute this action to the confusion that existed in the control room. 85

Pressurizer level began to decrease after the High Pressure Injection system was stopped because of the continuing decrease of Reactor Coolant System temperature. At 7-1/2 minutes into the incident, saturation pressure was reached in the Reactor Coolant System and boiling began. The void formation in the Reactor Coolant System caused expansion of the water and an increase in pressurizer level. At this point, the operators were still involved with responding to alarms and checking proper operation of systems. However, they began to realize that the plant was not responding as they had expected, particularly in light of the fact that pressure had continued to decrease. Some of the operators thought initially that this pressure decrease might be caused by overcooling of the Reactor Coolant System caused by the injection of cold water into the No. 1 steam generator, 86 however, others realized that they were losing Reactor Coolant System water. At approximately 9 minutes pressure stabilized at 980 psi and pressurizer level was offscale high. The operators found this combination very confusing, but they realized that the system was saturated, and that the pressure was remaining constant and the pressurizer level was high as a result of the boiling in the Reactor Coolant System. 87,88 At approximately 9 minutes 20 seconds, the operators tripped one Reactor Coolant Pump in each loop to reduce the heat input to the system. Only in retrospect did the operators realize that securing pumps to reduce heat input was not consistent with their concern that pressure decrease might be due to overcooling. 89

Reactor Coolant System pressure remained constant for approximately the next 13 minutes, while at the same time pressurizer level remained offscale high. At approximately 22 minutes, the operators received a high containment pressure alarm. This alarm, coupled with an instrument reading of 3 psig, caused one of the operators to finally realize that a leak was occurring from the Reactor Coolant System. This fact, as well as earlier information about the quench tank rupture disk blowing and other matters indicated to him that the PORV was open, and he immediately shut the block valve. 90 Shutting the block valve while the makeup pumps were running caused a repressurization of the system. This repressurization collapsed the steam bubbles that had formed in the Reactor

Coolant System, and pressurizer level rapidly decreased. Because of this decrease, the operators manually restarted the High Pressure Injection pumps.

Approximately 1 hour after the incident began, the operators had increased Reactor Coolant System pressure above saturation and had returned pressurizer level to normal. As a result, they secured the High Pressure Injection system a second time. At this point, the plant was in essentially a stable condition.

(a) Response to the Incident

(i) NRC Office of Inspection and Enforcement, Region III

The NRC Office of Inspection and Enforcement, Region III in Chicago was first notified of the incident by telephone at 8:45 a.m. on Sunday, September 25, 1977, the day following the event. The event was perceived by the Region III personnel as being a very severe transient, but, because the plant was in a safe condition, it was decided that it was not necessary to send someone to the site immediately. 91 The Principal Inspector for the Davis Besse plant, Thomas Tambling, was scheduled for a training session during the week following the incident. So another inspector, Terry Harpster was sent to the plant on Monday, September 26.

The purposes of Harpster's trip to the plant were to (1) determine if the plant was in a safe shutdown condition, (2) determine all the relevant parameters during the transient, (3) ensure that proper analysis of the transient was conducted, 92 and (4) define actions necessary before any further plant operation. 93 Harpster's review, which lasted approximately 1 week, raised several concerns that were subsequently related to Tambling. These concerns included (1) the operator response during the transient, (2) evaluation of the pressure excursion, including boiling effects in the core and the effects of boiling on the fuel, and (3) a possible problem with the High Pressure Injection system due to the fact that the operators were not sure if High Pressure Injection had gone into the core. 94

Harpster's concern about operator response centered on the fact that the operators had not had adequate training to recognize the problem with the Steam and Feedwater Rupture Control System, particularly because this system was unique to Davis Besse. Harpster was also concerned about the failure of the operators to integrate plant parameters (e.g., their reliance entirely on pressurizer level). However, he did not voice this second concern because the emphasis of his work and his major concerns were associated with plant physical problems. 95 Harpster also considered the generic implication of this incident; however, he thought it unreasonable to conclude that a similar transient could occur elsewhere because of the mechanical failures involved and the fact that the Steam and Feedwater Rupture Control System that initiated the incident was unique to Davis Besse. 96 Harpster was subsequently involved in a training session for various reactor inspectors and staff personnel at Region III. This session included a discussion of the chronology of events, the initiating sequence, the operator response, and the various equipment malfunctions.

On September 30, 1977, an Immediate Action Letter 97 was issued by Region III as a result of the September 24, 1977 incident. Among other things, this letter required an evaluation of the pressure excursion including boiling effects, to ensure that boiling did not cause damage to the Reactor Coolant System. I&E practice and policy required that this evaluation be completed before the plant was returned to Mode 4 (hot shutdown). 98,99

When Tambling assumed responsibility for the investigation, his primary concern was resolving specific items in the Immediate Action Letter. 100 Tambling was aware that void formation had occurred in the Reactor Coolant System, but he viewed it principally as a potential equipment problem associated with vibration of the Reactor Coolant Pumps and potential fuel damage. Tambling did not realize that void formation had caused the pressurizer level to increase; consequently, he believed that the operator action of securing High Pressure Injection was appropriate in view of the fact that pressurizer level had returned to the operating range. 101 Tambling also considered the generic implications of the incident. However, he concluded that no generic issues were associated with the incident because the Pilot Operated Relief Valve (PORV) that had failed open had been designed by one manufacturer, but the valve in other B&W plants was designed by a different manufacturer. 102 In addition, the fact that the relay in the PORV control circuit was missing was considered a plant problem and would not be expected to occur at other facilities. 103

At the conclusion of his inspection, Tambling requested that the licensee prepare a supplement to the initial Licensing Event Report (LER NP-32-77-16)(Ref. 104) that would include the analyses that Tambling had already reviewed at the site. This material (LER NP-32-77-16 Supplement)(Ref. 105) was forwarded to the Region III office on November 14, 1977, as a part of the report that is required within 90 days following such incidents.

The results of Tambling and Harpster's investigation were documented in an Inspection Report (No. 50-346/77-32) dated November 22, 1977 (Ref. 106). This report describes the incident as a sudden depressurization and notes several conclusions that are relevant to this Special Inquiry: (1) the operators had problems discovering that the PORV was open because of lack of direct indication of the valves position, and therefore, Toledo Edison installed indications of position of the PORV pilot valve, (2) the PORV control circuit was not safety-related and not covered by the quality assurance program for safety-related components, and (3) B&W had analyzed the incident and found that it was within the scope of the generalized depressurization transient previously analyzed. As a result of this inspection, no items of noncompliance associated with the incident were noted.

This concluded Region III involvement with this incident until concerns about this incident were raised by James Creswell, Region III inspector. These concerns are discussed in detail in Section IC11 of this report.

(ii) NRC Office of Nuclear Reactor Regulation

The NRC Office of Nuclear Reactor Regulation (NRR) also became involved with the investigation of this incident. Leon

Engle, the Licensing Project Manager for Davis Besse, was notified of the event by the Office of Inspection and Enforcement. However, because I&E did not request assistance, Engle concluded that active involvement by NRR was not required. 107 At the same time, the Division of Systems Safety within NRR also became aware of the event, and a fact-finding group headed by Gerald Mazetis was sent to the plant. Engle, Mazetis, and several other representatives of the Division of Systems Safety met with representatives of the utility, B&W, and Region III at the site on September 30, 1977.

Engle collected data from the incident and, after returning to Washington, plotted this data (see Figure 1(3) Although the data plots revealed that steam formation had caused the pressurizer level to increase, Engle did not consider this finding to be significant. He also realized that the operators had secured the High Pressure Injection system before isolating the leak. However, he did not focus on whether or not this action was proper because he considered operator action to be a responsibility of I&E. 108 His primary concern was the fact that a relay such as the one that was found missing in the PORV control circuit could be removed from a system without anyone's knowledge. 109 He believed that little action could result from this concern because the system was not considered to be a safety system. He was also concerned that the investigation was being conducted unsystematically because of the number of groups involved and the lack of coordination. He informed his supervisor of this concern, but nothing was done. 110

After his review, Mazetis prepared a handwritten Trip Report 111 in which he noted that saturation pressure was reached during the event and that the operators secured High Pressure Injection when they observed an increasing pressurizer level. In this informal report, he related several issues and concerns, including: (1) there were endless speculations associated with this event, and (2) the licensee should address the dynamic effect of vapor formation in the Reactor Coolant System during the transient, particularly because it was associated with Reactor Coolant Pump cavitation and seal effects. This informal report may not have been distributed to anyone. Mazetis has testified that he did not consider these concerns to be any more significant than other safety concerns that came up daily. 112

On October 3, 1977, Mazetis gave a briefing to representatives from the Division of Systems Safety and I&E including Roger Mattson, the Director of the Division of Systems Safety, and Karl Seyfrit, the Assistant Director, Division of Reactor Operations Inspection in I&E. The general characteristics of the transient were discussed, as was the plot of pressurizer level, Reactor Coolant System temperature, and Reactor Coolant System pressure prepared by Engle (Figure 1). The conclusion of this meeting was a decision by Seyfrit and Mattson that I&E would maintain lead responsibility for the investigation. 113

Subsequently, Mazetis prepared a note dated October 20, 1977, from Denwood Ross of NRR to Seyfrit. 114 The note described some areas of interest to the Division of System Safety that he believed should be addressed in the Toledo Edison Company formal report of the incident. One concern stated, "The operator's role in participating in the event should be related. For example, the manual actions associated

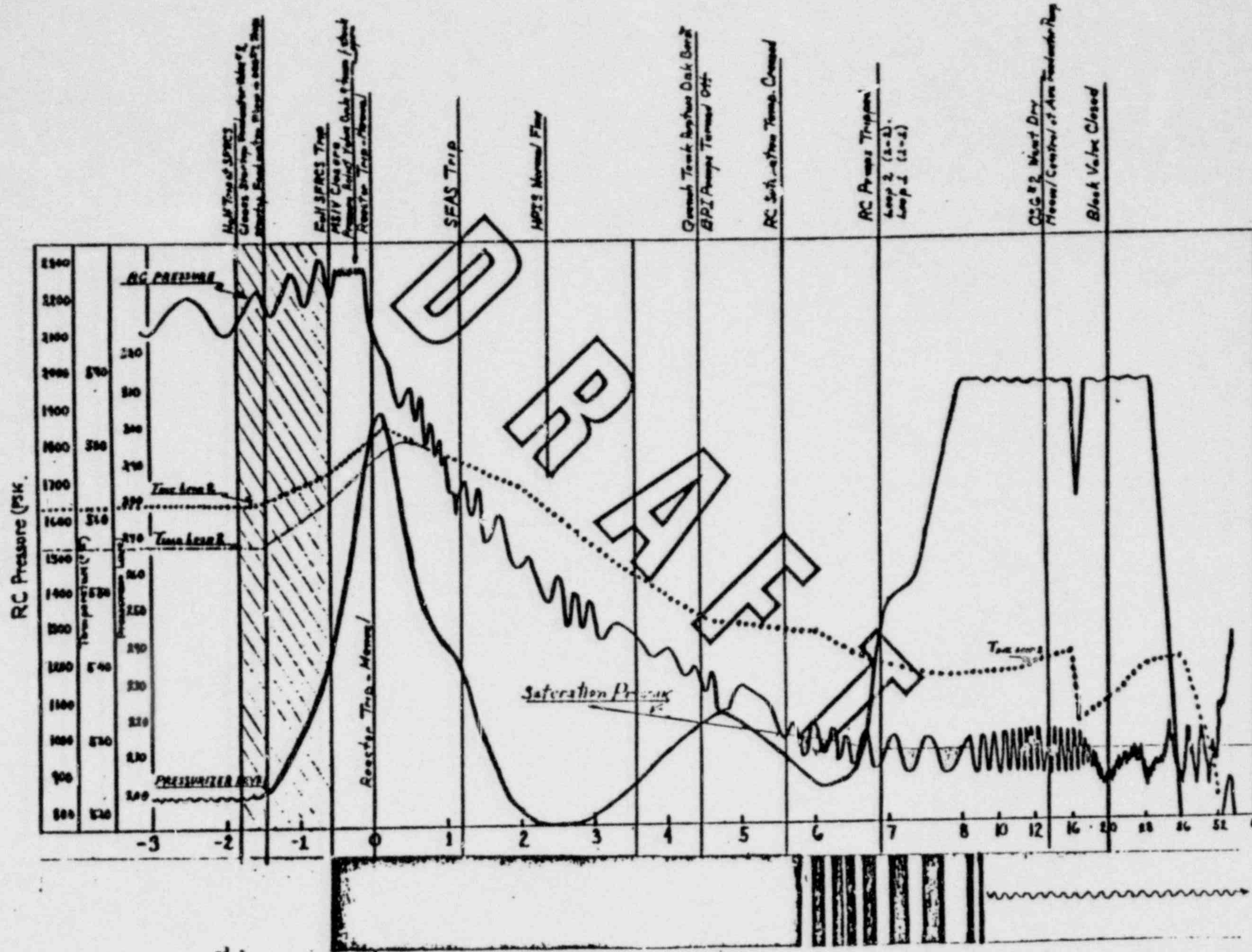

FIGURE IC 3

Davis Besse, Unit 1
Reactor Trip from ~9% Full Power at 2136 hours: September 24, 1977

with the control of level in steam generator No. 2 should be described. The operator's decision to secure high pressure injection flow based on pressurizer level indication should be explained." 115

Seyfrit does not recall whether he received this note; however, he believes that if he had received it, he would have called Region III or sent a copy of the report to the people conducting the investigation in Region III. 116 Testimony by Region III personnel and a review of the Region III files failed to produce the document or any recollection on the part of Region III personnel concerning the issues raised by this document. The meeting on October 3, 1977, and the October 20, 1977 note appear to be the only forums in which the concerns raised by NRR personnel would have been forwarded to the I&E inspectors conducting the investigation. The October 20, 1977 note apparently ended the Division of Systems Safety involvement.

R.J. McDermott of the Quality Assurance Branch in the Office of Nuclear Reactor Regulation also conducted a review to determine if deficiencies in the licensee's quality assurance program or test program had caused or contributed to the transient. In a memo dated October 6, 1977 (Ref 117), McDermott noted that the Emergency Core Cooling System had initiated at 1600 psig, that pressure reached as low as 800 psig, and that boiling occurred in the Reactor Coolant System. He did not comment on these facts. He noted that he did not have sufficient information to reach a conclusion, but that he had requested additional information from the I&E inspector. On October 20, 1977, McDermott wrote a memo 118 in which he concluded that the licensee had not been able to determine why or how the relay in the PORV control circuit was removed. This memo concluded McDermott's involvement. It does not appear that any subsequent actions were taken as a result of this review.

(iii) NRC Office of Inspection and Enforcement, Headquarters

In addition to the meeting on October 3, 1977, Karl Seyfrit participated in a briefing of the Advisory Committee on Reactor Safeguards (ACRS) on October 7, 1977 concerning this incident. During this briefing, it was noted that some boiling had occurred in the Reactor Coolant System. However, Seyfrit concluded that the transient was completely terminated after about 15 minutes by putting the No. 2 auxiliary feedwater pump in manual. 119 This was an interesting observation since the PORV was still stuck open at this time. Ebersole, who had already received the handwritten draft of the Michelson report (see Section IC7) and who subsequently prepared the Pebble Springs questions (see Section IC10) asked questions during this briefing. Specifically, he asked if High Pressure Injection had pumped water into the Reactor Coolant System. Seyfrit's response was that it had not because the operator had turned it off. 120 Ebersole also asked if it was planned to extrapolate the event to 100% power. Seyfrit stated that it was not likely that the plant could be in this particular position at 100% power. 121 Seyfrit's conclusion that the plant could not have a transient such as this at 100% power was based on the following points: (1) the plant was operating by dumping steam to the condenser rather than using the main turbines; (2) the plant was using the Startup Feedwater System rather than the Main Feedwater System (the spurious half-trip of the Steam and Feedwater Rupture Control System which initiated the incident would not have isolated feedwater flow to the No. 2 steam generator if the Main Feedwater System had been in use); and (3) different systems would be in operation and therefore would change the nature of the transient. 122

During the ACRS briefing, Seiss, a member of the ACRS, stated that Davis Besse had submitted what appeared to be an abnormally large number of Licensing Event Reports. He offered three hypotheses; (1) the number was, in fact, abnormally large for a plant startup, (2) the number was typical of plants during a startup, or (3) Davis Besse personnel had a different interpretation of what should be reported. Seyfrit stated that the answer was a combination of all three; but, he concluded that the performance at Davis Besse was not unique or unusual. 123

Seyfrit discussed this incident again at the November 1977 ACRS meeting. During the discussion, he noted that some cavitation had occurred in the Reactor Coolant Pumps due to boiling, but that no damage had occurred. Ebersole again asked about the implications of the same accident at full power. Seyfrit again responded that the same combination of events would be unlikely at full power. 124

(iv) B&W Response

Fred Faist is the Site Operations Manager for B&W at the Davis Besse plant. His initial involvement with the incident began with attendance at a meeting with Toledo Edison personnel at 10:00 a.m. Sunday morning (September 25, 1977). The purpose of the meeting was to identify the recovery effort that would be required and to review the sequence of events. 125

Faist subsequently requested that additional personnel be sent from the B&W offices in Lynchburg to support this effort. Therefore Joseph J. Kelly was sent to the Davis Besse plant to assist in the analysis of data that had been collected during the incident.

Kelly spent approximately 2 days at the plant, attempting to determine the sequence of events. Kelly did not consider what the operators saw or how they interpreted what they saw. His understanding was that the utility was interested primarily in assigning tasks to be accomplished before returning plant to service, and this was the emphasis of his work. 126 When Kelly returned to B&W, he gave a briefing in Lynchburg to people who were later sent to the plant to support Toledo Edison in its meeting with the NRC.

Kelly had identified several concerns that he raised with Faist and with B&W personnel in Lynchburg. These concerns included (1) fuel damage because of boiling in the core; (2) Reactor Coolant Pump damage resulting from operation at saturation conditions; (3) mechanical stress to the steam generators resulting from increased temperature difference associated with lost insulation; (4) chemical damage caused by boric acid crystallization on carbon steel pipe; (5) stress associated with excessive cooldown rates; and (6) the PORV failure. 127

During the briefing of B&W personnel in Lynchburg, Kelly discussed with Bert Dunn and Robert Jones of the B&W staff a concern associated with the steam formation in the Reactor Coolant System. Dunn resolved Kelly's concern about boiling and the possibility that it would damage the core, but raised a new concern about the operators incorrectly securing High Pressure Injection. 128,129 This led Kelly to prepare a memo concerning the guidance provided to operators associated with securing High Pressure Injection 130 (see Section IC9).

Faist also worked on the recovery effort following the incident. Some concerns that he identified include the following

(1) The alarm on one High Pressure Injection leg cleared, but the operators did not see flow indication in that leg. (Faist believes that this occurred when the operators manually initiated High Pressure Injection, 131 but others believed that this occurred when High Pressure Injection initiated automatically early in the incident.)

(2) Michael Derivan, the shift foreman in the control room during the incident, was confused by the fact that pressure decreased while pressurizer level increased. However, Faist testified that he did not consider the possibility that other operators might subsequently be confused. 132

Faist has testified that he had discussed the operation of High Pressure Injection during the incident with Dunn and Jones of B&W, and they concluded the High Pressure Injection should not have been turned off because of the possibility that it would not restart correctly if it were needed later in the incident. 133 However, it does not appear that Faist did anything as a result of this discussion.

Faist prepared a Site Problem Report (No. 372) (Ref. 134). He has testified that he tried to describe the hardware problems that had occurred and the sequence of events, as opposed to opinions and interviews with personnel. 135 Therefore, he did not record the fact that the operators were confused by the indication that they saw, nor did he report that the operators secured High Pressure Injection incorrectly. He simply noted that the operators had secured High Pressure Injection.

In the Site Problem Report, Faist also pointed out that the Steam and Feedwater Rupture Control System actuation did not trip the reactor. Toledo Edison opposed installing such a trip because they wanted to keep the Steam and Feedwater Rupture Control System and the Reactor Protection System separate. Toledo Edison personnel believed that the Reactor Protection System would trip the reactor when required. Faist did not consider the generic implications of the need for a similar anticipatory trip, based on loss of feedwater, on other B&W plants. 136

(v) Toledo Edison

The involvement of Toledo Edison management began during the actual incident. Terry D. Murray, the Assistant Station Superintendent (Murray became the Station Superintendent in November 1977) was at the plant when the incident occurred.

Murray arrived in the control room shortly after the operators manually tripped the reactor and he remained there throughout the incident. After Murray was confident that the plant was stabilized in a normal hot shutdown condition, he telephoned the station superintendent to inform him of the incident. 137 Murray did not contact the NRC at this time. 138

On Sunday morning (September 25, 1977) a meeting of station staff and support personnel was held to: 1) review the details of the incident, 2) identify issues that required additional investigation, and 3) develop a plan to correct physical damage that occurred inside the containment. 139 Shortly before the group convened, the NRC was contacted.

The principal concerns that came out of this in-house conference were: 1) potential damage to Reactor Coolant Pumps and to the fuel due to void formation in the Reactor Coolant System, 2) thermal stress of the Reactor Coolant System, 3) mechanical damage inside containment, and 4) the cause of the sticking of the PORV. 140

Two or three weeks after the initial meetings concerning the incident, the personnel who were in the control room met with a group of consultants to the president of Toledo Edison. During this conference the operators discussed the information available in the control room. 141,142 It was observed during the discussion that a common thread in these events was the operator's inability to recognize small LOCAs. 143 At least one of the operators also stated that his training had not prepared him for this event because he had never seen a leak where pressurizer level increased. 144 It does not appear that any actions were taken as a result of this meeting. In addition, this was the only time that the operators were asked to describe the difficulty they had in determining what was happening during the event. 145

(b) Specific Conclusions

(i) The incident that occurred at Davis Besse is almost an exact copy of the accident that subsequently occurred at TMI. The reasons that Davis Besse did not sustain the severe core damage that resulted at TMI are that (a) the Davis Besse plant had been operating at a very low power level and had a very low power history, and (b) the operators at Davis Besse were able to identify and isolate the open PORV in 20 minutes as opposed to 2 hours at TMI. If it had not been for these fortuitous conditions, it is very likely that the incident at Davis Besse would have been as severe as the subsequent accident at TMI-2.

(ii) Numerous groups were involved with the review of the incident at Davis Besse; a team from the Office of Nuclear Reactor Regulation, an individual from B&W in Lynchburg, two inspectors from the Office of Inspection and Enforcement, and plant personnel. Unfortunately, their efforts were not coordinated, and consequently the concerns raised by individuals were never exchanged among the members of the organizations. For example, the concerns raised by Mazetis in the Office of Nuclear Reactor Regulation that subsequently were forwarded to the Office of

Inspection and Enforcement as the Ross-Syfrit note were never forwarded to the I&E inspectors actually conducting the investigation. Similarly, the concerns raised by Kelly that subsequently resulted in the Kelly-Dunn memo were never forwarded to anyone outside of the B&W organization. Because of this fragmented investigation, there was never a cross-pollination of ideas, which might have resulted in a realization of the significance of some of the individual concerns.

(iii) All of the review groups overemphasized equipment. The reviewe ; tended to disregard the generic implications of the incident at Davis Besse by simply arguing that the specific pieces of hardware were different in other plants. This argument was proposed in spite of the fact that similar pieces of equipment with comparable probabilities of failure and similar failure modes were installed on other B&W plants and, in some cases, on all pressurized water reactors.

(iv) The people directly involved with the investigation made no significant effort to assess the scenario from the perspective of speculative analysis. Little consideration was given to what would have happened if the plant had been at a higher power level or a higher power history, or if it had taken the operators longer to identify an isolate the stuck open PORV.

(v) The information concerning the incident that occurred at Davis Besse was not effectively distributed to other B&W utilities, specifically to Metropolitan Edison. However, this is due primarily to the fact that the people directly involved with the investigation of the incident did not identify the significant issues associated with the incident that should have been identified, and they dismissed the generic implication of the incident by their emphasis on the equipment failures rather than an emphasis on the overall scenario that occurred.

(vi) In reviewing the incident at Davis Besse, one can see several indications that the PORV was open and that the Reactor Coolant System inventory was decreasing. With the benefit of hindsight the operators' actions appear to include a number of errors. These errors include stopping the High Pressure Injection pumps as the Reactor Coolant System approach saturation conditions and delay in closing the PORV block valve.

Study of the behavior of highly trained people under emergency conditions suggests that such people rarely make simple blunders in the operation of systems. Such people typically are highly disciplined; trained to follow procedures carefully; trained to avoid improvisation; and intensely aware of rules and constraints. Compared with the average person, they rarely make tactical errors in the sense of accidentally turning the wrong knob. Nevertheless, such trained people sometimes do make errors in emergencies. To distinguish these from the ordinary kind of errors, we may call these "strategic" errors. In an emergency such people recognize that something is wrong and that some action must be taken. They conceive a model or scenario for what is happening.

They follow procedures or reaction strategy which they believe is applicable to the scenario. Studies also show that once a scenario is conceived and a reaction strategy undertaken, there is a tendency not to seek or perceive additional data which contradict the original scenario. There is a psychological phenomenon called "cognitive dissonance" which makes the mind tend to reject data in conflict with the original hypothesis. 146

After an incorrect scenario is conceived, an entire pattern of actions can be taken which in retrospect are blunders. This phenomenon can be seen to a limited extent during the September 24, 1977 incident at Davis Besse, and to a much greater extent during the TMI accident. However, it does not appear that this phenomenon has ever been addressed in the design or licensing of nuclear power plants. The implications of this phenomenon are considerable since it implies that any sequence of actions by an operator, no matter how ill advised it may seem to a dispassionate observer, (i.e., the designer) may in fact be a creditable event that must be considered in accident analyses.

9. Kelly-Dunn Memoranda--November 1, 1977

Joseph Kelly of the B&W staff in Lynchburg, Virginia, was sent to the Davis Besse plant to assist Fred Faist, the B&W Site Operations Manager, in determining the sequence of events. Kelly's conclusions were given previously in section IC8(a)(iv).

Upon returning to Lynchburg, Kelly discussed the impact of steam formation in the Reactor Coolant System with Robert Jones (who subsequently became involved with the review of the Michelson report (see Section IC7)) and Bert Dunn of the B&W staff (see Fig. IC4 for the organizational relationships that existed). Dunn indicated that he did not consider that steam formation to be a particular problem, but, he did believe that the operators had terminated the High Pressure Injection system prematurely. He pointed out that he could develop scenarios in which the operators could have engendered serious consequences by securing High Pressure Injection when they did. 147,148

Kelly did nothing officially about Dunn's concern until he learned of a subsequent incident at Davis Besse on October 13, 1977 in which the operators prevented High Pressure Injection initiation. Because of this second example of what he considered to be improper operator action, Kelly wrote a memo dated November 1, 1977 (Ref. 149, 150).

Before writing this memo, Kelly talked to the simulator instructors at B&W and they stated that they did not understand why the operators reacted as they had. They stated that the operators had not been trained to secure High Pressure Injection unless Reactor Coolant System temperature had stabilized, Reactor Coolant System pressure was increasing, and pressurizer level was in the indicated band. 151

Kelly's November 1 memo noted that during the September 24, 1977 incident, "the operator stopped HPI when pressurizer level began to recover, without regard to primary pressure" 152 with should be pursued further. He has also agreed that the matter was not referred to DOR because no one considered whether the note should be sent to DOR.

The only case under active review where the note could have been applied prior to the TMI-2 accident was the Midland operating license application. However, requests for additional information sent to the applicant after the note was prepared do not include any questions that could have resulted from this note. The reviewer involved, Scott Newberry, testified that he does not know why the questions were not sent, although he does recall receiving the note. The only explanation that he can provide is that either (1) "it fell through the crack," possibly because it had to do with operating procedures which were not normally reviewed, or (2) he decided to wait until a later stage of the review process, possibly because the operating procedures had not yet been written for Midland. 323 Therefore, it appears that no action was taken with respect to the concerns described in this note, and that the material was never reviewed to determine if additional guidance should be provided to the licensees for plants already in operation.

Specific Conclusions

(a) We could not determine why Israel wrote the note. Apparently the reason was some combination of (i) the incident that occurred at Davis Besse on September 24, 1977; (ii) the handwritten draft copy of the Michelson report that was provided to Israel by Ebersole; or (iii) the questions that were asked during the ACRS review of the Pebble Springs Operating License application.

(b) The technical content of the Israel-Novak note did not describe the phenomenon that caused the reactor operators at Davis Besse, and subsequently at TMI, to secure High Pressure Injection. However, the note did describe a phenomenon that may have caused the pressurizer to remain full of water during the latter stages of the TMI accident when the Reactor Coolant System was essentially completely converted to steam.

(c) No actions were taken within the Reactor Systems Branch, the branch to which the note was addressed.

(d) The note was not sent to the Division of Operating Reactors for evaluation of its applicability to operating plants, apparently because of an oversight, rather than the result of any conscious decision not to send it.

13. Rancho Seco--March 20, 1978

On March 20, 1978, an incident occurred at the Rancho Seco* nuclear power plant when an operator dropped a light bulb into an instrument panel, shorting out a nonnuclear dc power supply. This short caused a reactor trip and a rapid cooldown at approximately 300 o F per hour. This rapid cooldown was greater than the cooldown rate limits permitted in the Technical Specifications for the plant. Furthermore, the loss of the dc power supply caused the loss of approximately two-thirds of the temperature, pressure, flow, and level signals available to the operator in the control room. During the incident, High Pressure Injection actuated at 1600 psig which maintained pressure above 1400 psig.

*Rancho Seco Nuclear Generating Station, which received an Operating License dated August 16, 1974, is owned by the Sacramento Municipal Utility District (SMUD). The plant is located 26 miles northeast of Stockton in Sacramento County, California. The reactor was obtained from Babcock & Wilcox Company and the plant uses a Westinghouse turbine generator. Bechtel Corporation served as the architect-engineer.

The event was reviewed by B&W and by the Sacramento Municipal Utility District (SMUD), and it was determined that the plant could return to power and that no significant damage had occurred. 324 However, the NRC staff noted that although no structural damage occurred, if the plant had operated for a longer time with the associated irradiation of the reactor vessel, more significant damage was possible as a result of brittle fracture associated with the rapid cooldown rate. The conclusions were that positive steps should be taken to prevent transients of this kind, and that the generic implications of the transient be promptly reviewed. This review was initiated in a memo from Darrell Eisenhut of the NRR staff to Victor Stello of the NRR staff dated March 30, 1978 (Ref. 325).

SMUD pointed out an additional problem, namely, that the incident had resulted in a loss of a significant amount of instrumentation and, consequently, the operators were hampered in their attempts to respond to the incident. This problem was caused not only by the erroneous indications observed by the operators, but also by the fact that the equipment responded in some cases to the erroneous signals that were received as a result of the loss of power. The operators found it difficult to determine which of their indicators were valid and which were incorrect. 326

This incident was also reviewed by I&E, and a formal Transfer of Lead Responsibility was executed on April 25, 1978 (Ref. 327), transferring responsibility for several issues from I&E to NRR. The issues raised in this transfer included: (1) review of the power supply to nonnuclear instrumentation to determine whether design changes were necessary; (2) review of the advisability of automatic initiation of auxiliary feedwater flow by a Safety Features Actuation System signal; and (3) evaluation of the susceptibility of B&W plants to other initiating events or failures that could produce similar cooldown transients. This Transfer of Lead Responsibility did not address the issue of the operator interpretation of indication or the availability of indication to the operators.

On June 20, 1978, a meeting held at Rancho Seco included representatives from NRR and from SMUD to discuss the cooldown transient. One purpose of the meeting was to determine whether other failures or initiating events could cause a similar transient. Conflicting reports exist concerning whether an additional failure mechanism was identified. One summary of the meeting indicated that none of the attendees postulated another mechanism or failure that would initiate a similar transient. 328 However, another summary of the same meeting stated, "The final item on the agenda was a discussion of other possible mechanisms for causing a severe cooldown transient. Depressurization due a faulty electromatic relief valve [i.e., PORV- or safety valve was the only possibility discussed." 329

Regardless of what was actually decided at the meeting, because of perceived higher priority work, further action on this entire issue was suspended after this meeting, and no additional actions were taken on any of the issues addressed in the Transfer of Lead Responsibility. 330,331

As already noted, B&W had also reviewed this incident and, on August 8, 1978, sent a letter to each of the Site Operations Managers (except at TMI-2) for subsequent forwarding to B&W plants. This letter discussed the severe thermal transient that had occurred at Rancho Seco and also discussed the substantial loss of nonnuclear instrumentation associated with the loss of electrical power. The letter observed further that need for a careful evaluation of operator training and emergency operating procedures for any loss of nonnuclear instrumentation. The letter emphasized that the operator's response should be keyed to certain variables if a loss of normally available instrumentation occurs. The specific variables cited as significant were (1) pressurizer level, (2) Reactor Coolant System pressure, (3) steam generator level, and (4) steam generator pressure. The letter stated, "The pressurizer level and reactor coolant system pressure assure that the reactor coolant system is filled; the steam generator level and pressure assure adequate decay heat removal." 332

As stated earlier, this letter was sent to all B&W utilities except Metropolitan Edison, the operator of TMI-2. The reason this letter was not sent to TMI is that an earlier incident had occurred at TMI on April 23, 1978, and it was thought by B&W that this issue had been discussed with TMI in sufficient detail that it was not necessary to send them the letter. However, no specific documentation concerning these discussions was found. Another reason for not sending the letter to TMI was that the TMI Integrated Control System involved in the response to the erroneous indication was different from the system installed at Rancho Seco. 333 If this letter had been sent to TMI-2 it might have resulted in operator training that emphasized the need to consider Reactor Coolant System pressure, and not just pressurizer level, when attempting to determine Reactor Coolant System inventory.

DRAFT

Specific Conclusions

(a) The incident itself was not a direct precursor of the TMI-2 accident (i.e., the incidents themselves are not similar).
(b) A letter was prepared and forwarded to various B&W utilities. The letter discussed the fact that Reactor Coolant System pressure and pressurizer level were the measures of Reactor Coolant System inventory. Had TMI-2 received this letter, it might have resulted in additional emphasis and training at TMI-2 with respect to the fact that pressurizer level alone was not an accurate indication of Reactor Coolant System inventory. The letter was not forwarded to Metropolitan Edison, however, because B&W concluded that the issues contained in the letter had been discussed with them during the review of a similar incident which had occurred at Three Mile Island on April 23, 1978. This discussion is not, however, a matter of record at either B&W or Toledo Edison.

14. Three Mile Island--March 29, 1978/Sternberg Memo--March 31, 1978

On March 29, 1978, a reactor trip occurred at TMI-2 as a result of the loss of a vital bus. Power to the vital bus was lost because of the tripping of the alternative power supply during a test. This loss of power caused the PORV to fail open on loss of power to the control bistable, causing a depressurization of the Reactor Coolant System. Furthermore, the High Pressure Injection system initiated. The depressurization was stopped after about 4 minutes by reenergizing the vital bus from its alternate power supply.

The utility noted that there was a problem associated with this incident because the PORV opened (rather than closed) on loss of power to its control bistable. In a Startup Problem Report dated March 30, 1978 (Ref. 334), the utility suggested either changing the valve to fail shut or providing an indication on the control panel that the valve had an open signal.

This matter was reviewed by B&W and the conclusions were that (1) B&W agreed with the concept of having the valve fail shut on loss of nonnuclear instrumentation, and (2) the indication of the PORV position should be provided in the control room; however, this indication was to come from the power to the solenoid. 335

This issue was also reviewed by the architect-engineer, and an Engineering Change Memo was initiated on April 6, 1978 (Ref. 336). The Engineering Change Memo provided for an indication in the control room of power to the solenoid. The memo initially included a provision for changing the PORV to fail shut on loss of power; however, that provision may have been subsequently deleted because it was not required for proper system operation. 337 Whether the PORV was eventually changed to fail shut on loss of control power was not determined. Burns and Roe also concluded that, even though it would require a change to the Final Safety Analysis Report, the change was not an unreviewed safety question. 338

These actions were subsequently reported to the I&E Region I office by Metropolitan Edison in a letter dated June 27, 1978 (Ref. 339). This letter concludes that Reactor Coolant System pressure reached as low as 1173 psig during the event and that (1) the control signal should be changed to cause the valve to fail shut on loss of control power, and (2) position indication for the PORV should be provided in the control room.

During this period, Daniel Sternberg of the I&E Region I office also became concerned as a result of this incident. Sternberg was the Acting Branch Chief for the I&E branch responsible for TMI-2. He prepared a memo to I&E Headquarters, dated March 31, 1978 (Ref. 340), in which he noted that the March 29, 1978 incident resulted in a blowdown because the PORV opened on a loss of electrical power to the control bistable. Although Sternberg acknowledged that the valve was not safety-related, he stated,

"It is requested that the adequacy of the design approach (i.e., valve failing open on loss of control power) be reviewed on an expedited basis for B&W facilities in general and Three Mile Island in particular." 341

Sternberg has testified that he was concerned because the PORV failed open on the loss of a single power supply, and this failure resulted in an initiation of an unannunciated loss-of-coolant accident. 342 Sternberg believed that his ability to correct problems such as this was significantly impaired since the item was not defined as a safety-related component. Nonetheless he thought that the issue should be addressed. He also testified that he would have recommended that the matter be referred to NRR for review, but he had been told earlier in his career in I&E Region I not to make such recommendations because such decisions were the prerogative of I&E Headquarters. 344

Sternberg received a response from I&E Headquarters signed by Karl Seyfrit on May 3, 1978 (Ref. 345). The response, which was prepared by Roger Woodruff, stated,

> "The request is based on failure of the valve in the open position. Failure in this position is covered in Section 7.4.1.1.6 of the FSAR. We conclude that additional review is not warranted." 346

Section 7.4.1.1.6 of the FSAR, titled "Pressurizer Control," states, "In the event that the relief valve were to fail in the open position, pressure relief could be controlled by cycling (open and close) the relief isolation valve." 347

Woodruff did not contact anyone in NRR about this matter because he thought that the issue had already been reviewed by NRR. Furthermore, he did not think the valve should be safety-related because the code safety valves, which provide relief protection if the PORV fails to open, are safety-related. 348

Sternberg has testified that he accepted the response as adequate because someone had reviewed the issue and decided that it was not a problem. However, he would have preferred to see an analysis of the implications of a valve that can cause a small loss-of-coolant accident by failing open on a loss of control power. Because of perceived higher priority work, however, Mr. Sternberg did not pursue the issue after he received the memo from I&E Headquarters. 349

Although Seyfrit did not personally review the matter in detail, he thought that because the issue was addressed as part of the application, and that application had been reviewed by NRR previously, the design was acceptable. 350

Specific Conclusions

(a) The memo is a precursor to the TMI-2 accident because it refers to an incident that occurred at TMI (March 29, 1978) during which a PORV failed in the open position creating a small LOCA. Although this failure, was due to a loss of control power, it had the same effect as the failure, for whatever reason, a year later.

(b) A re-examination by NRR of the adequacy of the design of the TMI-2 PORV, might have precipitated an assessment of the implication of a stuck open PORV, or might have provided the impetus for an adequate PORV position indication in the control room. Such a reexamination never occurred.

NUREG-0572

Review of Licensee Event Reports (1976 - 1978)

Advisory Committee on Reactor Safeguards

U.S. Nuclear Regulatory Commission