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EVALUATION OF INCIDENTS
OF PRIMARY COOLANT RELEASE
FROM OPERATING BOILING WATER REACTORS

RETURN TO REGULATORY CENTRAL FILES
ROOM 016

Office of Operations Evaluation
Directorate of Regulatory Operations

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IMPLEMENTATION OF RECOMMENDATIONS OF THE REGULATORY STUDY GROUP

A regulatory study group was appointed in May 1972 to evaluate incidents of coolant release at operating boiling water reactors. The final report of the study group has been completed.

The recommendations contained in the report have been considered by the Director of Regulation and are to be implemented by the appropriate Regulatory Directorates.

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SUMMARY

At the request of the Director of Regulation, the Office of Operations Evaluation conducted an evaluation of incidents and unusual occurrences in operating boiling water nuclear power plants involving the inadvertent release of reactor coolant from the primary coolant pressure boundary. The scope of the study and the broad conclusions and recommendations reached by the study group assigned to conduct the evaluation are summarized herein.

The scope of the study was limited to eight incidents involving the unintentional discharge of significant quantities of primary coolant during reactor operation. All eight of the incidents involved Boiling Water Reactors (BWR). The significance of these incidents with respect to the health and safety of plant operating personnel and the public was negligible. The exposure to radiation of onsite personnel and the concentration levels of radioactive effluents released beyond the site boundary were well below the permissible levels contained in 10 CFR 20.

No two of the incidents were initiated by a common system or component malfunction. In each instance, malfunctions of equipment resulted in a reactor scram and turbine trip followed by reactor vessel water level and primary system pressure transients characteristic of BWR's. The reactor primary coolant was released during these transients through safety and relief valves that either operated prematurely or that operated correctly but failed to close.

The limited operating statistics available indicate that approximately one coolant release incident attributable to relief or safety valve malfunction has occurred for each two reactor-years of operation.

Three passive barriers are provided in nuclear power plants to preclude the release of fission products. They are the fuel cladding, the primary coolant pressure boundary and the containment building. The safety significance of the incidents studied is thus related to the frequency that one of these barriers, the primary coolant pressure boundary, was violated and to the added safety burden placed on the two remaining barriers, particularly the reactor containment building. It has been concluded, therefore, that attention should be given to determining the basic cause(s) of safety valve malfunctions,

and to the timely development of plans to prevent or minimize occurrences of coolant release. If incidents of coolant release cannot be prevented or their frequency significantly reduced, the capability of the containment (drywell) and the safety equipment located therein to perform their intended functions throughout the life of the plant warrants further attention. This conclusion was reached with full recognition that no release of significant quantities of radioactive materials to the public domain has occurred as a result of these eight incidents.

The study revealed that the performance of the feedwater control systems for controlling water level automatically within acceptable bounds during unusual occurrences has been less than satisfactory. Although the performance of the feedwater control system has no direct safety significance, the burden placed on the reactor operator by the reactor vessel water level transients coupled with the potential effect of steam line flooding on safety valve actuation and on the operation of the High Pressure Coolant Injection System has led to the conclusion that the performance objectives of feedwater control systems in terms of maximum and minimum reactor vessel water level should be specified and that the system performance should be verified by appropriate startup test programs.

The actions of operating personnel during the incidents tended to aggravate and prolong several of the incidents. The cause for the operating errors and deviations from operating procedures experienced during the incidents could not be determined. Although the consequences of these errors and deviations were not significant for the incidents studied, collectively they may reflect shortcomings in one or more areas such as the control room design, operating procedures, and personnel training. It has been concluded, therefore, that a standard or guide should be developed by industry, with AEC participation, for control-panel and control-room design that better addresses the human engineering aspects of reactor operation during abnormal operating occurrences. Also additional emphasis should be devoted by licensees in the course of the plant safety analysis to evaluating the adequacy of the plant operating procedures for coping with anticipated transients.

The documentation available in the regulatory files revealed wide variations in the scope and depth of information reported by licensees regarding incidents and unusual occurrences. Sufficient documentation also was not available to determine whether all licensees of operating reactors receive timely reports of incidents at other similar

nuclear plants. In addition, it could not be determined whether licensees had evaluated the potential for similar incidents in their own plants and where appropriate, had taken corrective action to minimize the probability of similar occurrences. Because of these shortcomings in the documentation, it has been recommended that the Regulatory staff develop an incident reporting guide to supplement Safety Guide 16.C.2.a. It is further recommended that a system be developed and implemented by AEC to rapidly and fully inform all licensees of incidents or unusual occurrences. Further, inspections now conducted by Regulatory Operations field personnel should routinely include review of licensees' plans for evaluating and taking corrective action to prevent similar occurrences.

The scope of the Regulatory staff review of the incidents and the basis for the decisions made could not be fully determined from documentation available in the Regulatory files. Follow-up actions by the staff with respect to licensee commitments for conducting long term programs also were not generally available. These observations have led to the conclusion that Regulatory policies and procedures should be revised to more clearly identify the responsibility for review, decision making, investigation and documentation with respect to incidents and unusual occurrences.

1.0 Introduction

1.1 Purpose

On May 10, 1972, at the request of the Director of Regulation, the Office of Operations Evaluation of the Directorate of Regulatory Operations, U. S. Atomic Energy Commission, initiated an evaluation of incidents and unusual occurrences in operating light water-cooled nuclear power plants involving the inadvertent release of reactor coolant from the primary coolant pressure boundary. A special study group was appointed to conduct this evaluation.* The purpose of this evaluation was to assess the significance of such incidents to public health and safety and to appraise the programs for preventing or minimizing recurrence.

Specific objectives of the evaluation were:

1. To determine whether the initiating causes of the incidents or the subsequent plant responses and personnel actions revealed common characteristics attributable to specific systems or components;

2. to evaluate the adequacy and effectiveness of corrective actions taken and the sufficiency of on-going programs to reduce the probability of such incidents and mitigate their consequences; and

3. to assess the safety significance of the incidents on the future operation of nuclear plants.

The results of the evaluation are presented in this report.

1.2 Scope of Evaluation

The scope of this evaluation has been limited to incidents involving the unintentional discharge of significant quantities of coolant from the primary coolant pressure boundary of operating water-cooled nuclear power plants.

Since 1967, licensees of nuclear plants have reported the inadvertent release or leakage of primary coolant during plant operations on more than 50 occasions. The majority of these incidents involved small quantities of coolant and were caused by minor malfunctions of such things as gaskets, packing glands, pump seals,

*The membership of the study group is listed in Appendix A.

leakage through valve seats and minute cracks in piping. These incidents resulted in little or no damage to other equipment and presented no safety hazard to operating personnel or the public. All of these incidents or unusual occurrences were investigated by the regulatory staff at the time of occurrence and an evaluation made of corrective actions taken by the licensee and of the impact on continued operation of the plant.

The incidents reported since 1967 have been screened, based on the quantity of primary coolant released during the incident, the number and safety significance of equipment malfunctions or operator errors that occurred, and the severity of the damage to plant equipment. Based on these general criteria, eight incidents were identified during the screening process for consideration as part of this evaluation. A tabulation of these eight incidents, including the plant name and date of the incident, is presented in Table I. All eight of the incidents involved Boiling Water Reactors (BWR's). No incidents involving Pressurized Water Reactors (PWR's) were selected for evaluation because, in the judgement of the study group, no significant incidents of primary coolant release have occurred during reactor operation.

1.3 Approach

Evaluation of the incidents listed in Table I was based primarily on information in existing documents and correspondence in docket files of the Directorates of Licensing and Regulatory Operations and in published Reactor Operating Experience Reports. A list of the documents used in the study is included as Appendix E. Independent calculations were not performed to verify data reported. However, during its review and evaluation, meetings were held to clarify existing information with representatives of General Electric Company, Commonwealth Edison Company, Niagara Mohawk Power Company, Dresser Industries, Target Rock, Inc., and members of the regulatory staff. Discussions were held with several other licensee representatives. Some new information concerning on-going test programs was obtained during discussions and correspondence with Commonwealth Edison Company and General Electric Company. Members of the study group also visited the Boiling Water Reactor Simulator located near Dresden Station for the purpose of witnessing a control room simulation of the Dresden 2 incident of June 5, 1970, and the Dresden 3 incident of December 8, 1971.

The approach followed in carrying out this review was to evaluate each incident with respect to the following: 1) The initiating mechanism; 2) the sequence of events, performance of systems and

TABLE I

Incidents Considered for Evaluation

	Plant Name	Date of Incident
1.	Dresden - 2 Grundy County, Illinois Commonwealth Edison Co.	June 5, 1970
2.	Monticello Monticello, Minnesota Northern States Power Co.	September 28, 1971
3.	Millstone - 1 New London, Conn. Millstone Point Company	October 10, 1971
4.	Nine Mile Point - 1 Oswego, New York Niagara Mohawk Power Corp.	December 31, 1971
5.	Dresden - 3 Grundy County, Illinois Commonwealth Edison Co.	December 8, 1971
6.	Nine Mile Point - 1 Oswego, New York Niagara Mohawk Power Corp.	February 28, 1972
7.	Monticello Monticello, Minn. Northern States Power Co.	February 26, 1972
8.	Dresden -3 Grundy County, Illinois Commonwealth Edison Co.	May 4, 1972

The scope of the evaluation was limited principally, at this time, to the incidents occurring at Dresden station, Units 2 and 3.

components and operator actions during the incident; 3) the damage to plant equipment; 4) the corrective actions taken by the licensee; 5) the supporting documentation, and 6) the significance of the incident with respect to public health and safety. Summary descriptions of each incident are presented in Section 2.0. A detailed chronology of events for each incident is included as Appendix B. The reviews and evaluations were performed using a standard format to enable a consistent and uniform evaluation of all incidents.

In considering each incident, the design criteria and objectives for the systems and components involved were identified. The performance of these systems and components during the incident were then compared with the design objectives. The actions of the operator during the incidents were evaluated against the operating procedures.

The incidents were then evaluated as a group to assist in the identification of common characteristics and contributing factors to determine whether incidents of the type considered were more significant to safety than the independent evaluation of each occurrence might otherwise indicate. The results of the overall evaluation are summarized in Section 3.0 and supplemented by Appendix C and Appendix D.

The conclusions and recommendations of the study group presented in Section 4.0 represent the judgment of those members who were most familiar with the subject in question. Because members of the study group represent expertise in various disciplines, not all members contributed to all areas covered in this report.

2.0 Description of Incidents Considered

Brief narrative descriptions of each of the incidents considered by the study group are presented in this section. Background information concerning each of the nuclear plants involved and a summary of the damage to plant equipment and corrective actions taken by the licensee are also included. More detailed information concerning the chronological sequence of events is presented in Appendix A to supplement the brief descriptive summaries herein.

2.1 Dresden Unit 2, June 5, 1970

On June 5, 1970, an incident occurred at the Dresden Unit 2 nuclear plant following an unexpected reactor scram in which the discharge from a safety valve impinged on two other safety valves, causing them to open and to fail to close thereby releasing primary coolant to the containment. Background information concerning the Dresden Unit 2 is as follows:

Background Information

Name of Facility: Dresden Unit 2

Size and Type: 809 MWe; BWR

Location: Near Morris, Illinois

Owner: Commonwealth Edison Company

Architect-Engineer: Sargent and Lundy Company

Type of Contract: Turnkey

Date License Issued: December 22, 1969

While operating at a power of 75% (623 MWe) under near equilibrium conditions, a spurious signal in the turbine pressure control system occurred causing the turbine control valves to open from 75% to 80% (the load reference setpoint) and the steam bypass valves to open fully to the condenser. A turbine trip occurred, resulting in a signal causing the reactor to shut down (scram).

Subsequently, several transients were experienced in the reactor vessel water level and in reactor vessel pressure. Coincidentally, the pen on the reactor vessel water level recorder incorrectly indicated a low water level in the reactor vessel, and in an attempt to control the water level and pressure transients, the reactor operator chose to manually control the water makeup rate (Feedwater System) and to control reactor system pressure by operation of the relief valves. An excessive amount of water was added to the reactor vessel resulting in flooding of the steam lines. Operation of the pressure relief valve resulted in at least one safety valve opening below its setpoint. The discharge from this valve impinged on two other safety valves causing their lifting levers to lock in a partially open position. These two valves remained open for the duration of the depressurization and cooldown of the primary system.

The water-steam mixture discharging from the two safety valves caused an increase in the primary containment vessel (drywell) pressure. At 2 psig a drywell pressure signal initiated the Emergency Core Cooling system (ECCS), isolated the Reactor Building Ventilation System, started the Standby Gas Treatment System (SGTS) and started the Standby Diesel Generators. The High Pressure Coolant Injection System (HPCI) was out of service for repair during the incident, but would not have operated in any event because of high reactor water level in the reactor vessel. Both the Core Spray System and the Low Pressure Coolant Injection System (LPCI) started automatically in accordance with design objectives, but did not inject water into the reactor vessel because the pressure in the reactor vessel remained above the 350 psig interlock setpoint.

During the period of 30 to 40 minutes after initiation of the incident, the operator was unable to determine the value of the pressure in the containment, since the only pressure indicator (-5 to +5 psig) was off-scale. The operating staff supervisor made the decision not to operate the containment sprays, as procedures required, but to vent the containment via the Standby Gas Treatment System. Venting of the containment was controlled and monitored.

The liquid in the drywell was pumped to the Radioactive Waste System to prevent contaminated water from flowing into the torus.

Throughout the incident, radioactivity levels were monitored both in the plant and off-site. Gas discharges through the stack were monitored. The incident did not result in any measurable release of either liquid or gaseous radioactive products to the environment.

Equipment malfunctions that contributed to the incident included the following: 1) malfunction of the pressure regulator in the pressure controller, 2) sticking of the pen on the principal reactor vessel water level recorder, 3) improper operation of the isolation condenser, and 4) premature opening of a safety valve.

The damage to plant equipment was confined to the following: blistering and peeling of paint in the drywell in the vicinity of the safety valves that opened, damage to the Local Power Range Monitors (LPRM) Source Range Monitor levers on two safety valves, deterioration of thermal insulation on piping and components in the vicinity of the safety valves, and deterioration of electrical insulation of one drywell cooling blower motor.

Corrective action taken by the licensee consisted of the following: major revision of the operating procedures, particularly those involving feedwater control during transients; replacement of the LPRM, SRM and IRM cables; repotting of the containment electrical penetration seals; repainting of portions of the drywell and the vent piping, all safety valves were replaced with steam tested valves and lifting levers removed; reorientation of the safety valve discharge nozzles to preclude direct impingement on adjacent valves and components; replacement of one drywell cooling blower motor; replacement of the wire wound potentiometer in the pressure control unit; separation of signal and power cables for this electro-hydraulic control unit; increasing the setpoints for isolation trips on the isolation condenser to agree with the original design objective, and modification of the feedwater control system to minimize control valve leakage, improve the response of the flow control and minimize the probability of malfunction of the feedwater pumps because of low net positive suction head.

2.2 Monticello, September 28, 1971

On September 28, 1972, following a reactor scram, a relief valve was opened by the operator and failed to reseat, thereby releasing primary coolant water to the torus.

Background information concerning the Monticello Plant is as follows:

Facility:	Monticello I
Size and Type:	545 MWe - BWR
Location:	Monticello, Minn., 30 miles NW of Minneapolis, Minn.
Utility:	Northern States Power Co.

NSSS and Contract Type:	General Electric - Turnkey
Architect-Engineer:	Bechtel Corporation
Date of Operating License:	September 8, 1970

At approximately 1:25 a.m. on September 28, 1971, a malfunction in the recirculation pump control system produced a reactor scram and isolation of the steam supply system. The "D" relief valve (one of four relief valves in the plant) was manually actuated, in accordance with plant operating procedures, to maintain reactor pressure between 950 psig and 1050 psig by discharging steam to the torus. On the second manual actuation of the "D" relief valve, the valve failed to reseat after the operator returned the control switch to the closed position. Reactor pressure continued to drop until the "D" relief valve was given another open and close signal at which time the valve reseated.

The duration of blowdown was estimated to be 37 seconds during which time reactor pressure dropped from 1040 psig to 670 psig. No significant damage or releases of radioactivity were reported as a result of the event and the reactor went critical again at 10:58 p.m. the same day. Corrective actions taken as a result of the incident were: (1) testing of the manual control circuit for the "D" relief valve, (2) replacement of the main valve seats and discs, installation of stellite coated stems, chamfering of the valve stem bushings, and removal of the thermal insulation from all four relief valves, and (3) testing and adjustment of Recirculation Pump Controls.

2.3 Millstone Unit 1, October 10, 1971

On October 10, 1971 an incident occurred at the Millstone Unit 1 nuclear plant in which a relief valve opened and failed to reseat, thereby releasing primary coolant water to the torus. Background information concerning this plant is as follows:

Background Information

Name of Facility:	Millstone Point Company, Unit #1
Size and Type:	652 MWe - BWR
Location:	New London Co., Connecticut

Utility:	Northeast Utilities Company
NSSS and Type of Contract:	General Electric Company - Turnkey
Architect Engineer:	EBASCO Services
Date of Operating License:	October 7, 1970

On October 10, 1971, while operating at 100% power, a malfunction in the turbine pressure control system caused two steam by-pass valves to open. Turbine trip and reactor shutdown ensued. The opening of these by-pass valves and subsequent events resulted in a pressure increase in the primary system and the actuation of a relief valve. The relief valve failed to reseat at the proper pressure, thereby causing a release of primary coolant water to the torus.

There was no measurable release of radioactivity to the environment and no damage of components and structure attributable to the coolant discharge. As a result of this incident, and subsequent investigative and corrective actions, the plant did not operate for thirteen (13) days.

The corrective actions consisted of: (1) installing a new type of set pressure adjustment springs in all electromatic relief valves; (2) lapping of relief valve seats and discs as needed; (3) checking the lift pressure of all relief valves; and (4) modifying the turbine control system.

2.4 Nine Mile Point, December 1971

On December 31, 1971 an incident occurred in which water overflowed into the main steam lines and relief valves actuated, thereby releasing primary coolant water to the torus.

Background information concerning the Nine Mile Point One Plant is as follows:

Background Information

Name of Facility:	Nine Mile Point - 1
Size and Type:	625 MWe - BWR
Location:	Oswego, New York

Utility:	Niagara Mohawk Power Corporation
NSSS Contractor and Type of Contract	GE, Not Turnkey
Architect Engineer:	Niagara Mohawk Power Corporation
Date of Operating License:	August 22, 1969

On December 31, 1971, the power level of the reactor was 1752 megawatts thermal (95% of rated power), reactor pressure was 1015 psig, and steam flow was 6.8 million pounds per hour. The radioactivity release rates were averaging 25,000 microcuries per second at the time of the incident and the plant was being operated slightly below the rated power level because of operational difficulties with the turbine second stage reheaters. Routine surveillance testing of the reactor protection system high/low water level system was being conducted.

Bumping of sensor supports by a technician caused the high water level sensor to trip, resulting in a turbine trip. A reactor scram resulted from turbine trip. After about 20 seconds, the operator switched feedwater control to manual mode because, in his opinion, feedwater flow to the reactor vessel was high. Manual actuation was too slow and water overflowed into the main steam lines. Several operations of the electromatic relief valves occurred for approximately 12 minutes, after which reactor water level was brought under control. The emergency condenser was then placed in service to control reactor pressure. Following the turbine trip and until main steam line isolation occurred (about 28 seconds), the radioactive off-gas release rates increased to about 60,000 microcuries per second. When the system was returned to hot standby conditions, the off-gas was 40,000 microcuries per second and was reduced to about 15,000 microcuries per second when the system was returned to power operation.

No damage was reported as a result of the incident. The total down time was 13 hours 47 minutes. The procedure for operation of the feedwater system was reviewed with the operator. No modifications or repairs were reported.

2.5 Dresden Unit 3, December 8, 1971

On December 8, 1971, an incident occurred at the Dresden 3 Nuclear Power Plant. Background information concerning the plant is as follows:

Background Information

Name of Facility:	Dresden Unit 3
Size and Type:	809 MWe - BWR
Location:	Grundy County, Illinois
Owner:	Commonwealth Edison Company
NSSS Contractor & Type:	General Electric - Turnkey
Architect-Engineer:	Sargent and Lundy Company
Date of Operating License:	January 1971

The plant was operating in a base loaded condition at 792 MWe. A test was in progress to determine the quantity of fission product plateout on the chimney monitor sampling system. The chimney off-gas release rate was approximately 3,090 μCi/sec.

One of three operating condensate booster pumps tripped causing the two feedwater pumps to trip because of low suction pressure. The standby feedwater pump started automatically and provided feedwater at approximately one half the previous rate. The reactor scrammed on a low reactor vessel water level signal. Primary system pressure decreased, resulting in the automatic isolation of the main steam lines.

The increased demand for additional water caused the feedwater control system to revert to the flow control mode. The water level reached a minimum of -20" (123" above the active core) before starting to rise. The operator placed the feedwater control system into the manual control mode. The feedwater control valve failed to receive sufficient air and locked in its existing position. The operator was able to reduce, but not shut off, the feedwater flow rate by operating the feedwater control isolation valve. This valve stalled due to the high differential pressure across the valve gate. The water level in the reactor vessel increased to a level that flooded the main steam lines.

Approximately 5 minutes after the scram, a safety valve located on a main steam line lifted for approximately 1.5 minutes at a system pressure about 200 psig below its setpoint. The containment was

pressurized to a maximum of 20 psig and reached a temperature of 295°F. The ECCS systems were automatically activated but were not called upon to function.

The operating feedwater pump was manually tripped and the transient terminated. Operation of the pressure suppression system was used to reduce the containment pressure. Containment integrity was maintained and there was no release of radioactivity to the environment.

The incident resulted in damage to: LPRM cabling, a containment cooling fan motor, a solenoid on an electromatic relief valve in the 125 volt d.c. system.

The reactor was shut down for 20 days during which time all of the above damaged items were replaced or repaired. In addition, the feedwater control valve air system capacity was increased to preclude malfunctions due to lack of sufficient air. The condensate booster pump was disassembled and inspected and all circuits tested. Several changes also were made to operating procedures involving control of feedwater system transients. In addition, programs were initiated to investigate feedwater control system and safety valve performance.

2.6 Nine Mile Point, February 28, 1972

On February 28, 1972, an incident occurred in the Nine Mile Point nuclear plant in which water overflowed into the main steam lines and relief valves actuated, thereby releasing primary coolant water to the pressure suppression pool. Background information concerning the Nine Mile Point One Plant is as follows:

Background Information

Name of Facility:	Nine Mile Point One
Size and Type:	1850 MWt - BWR
Location:	Oswego, New York
Utility:	Niagara Mohawk Power Corporation
NSSS Contractor and Type of Contract:	GE, Not Turnkey
Architect Engineer:	Niagara Mohawk
Date of Operating License:	August 22, 1969

On February 28, 1972, the power level was 1760 megawatts thermal (95% of rated power), reactor pressure was 1011 psi and steam flow was 6.25 million pounds per hour.

A malfunction in the control circuit of a continuous motor-generator (M/G) set power supply caused the motor-generator (M/G) set to coast down, resulting in a loss of generator output voltage. This caused a 1/2 scram because of a trip in all sensors in only one channel of the reactor protection system. The loss of generator output voltage caused loss of power to parts of the feedwater system which caused feedwater upset, control valve lockup and subsequent reactor scram due to low water level. The loss of generator output voltage also caused loss of the cleanup system. The main steam isolation valve closed when the pressure decreased to less than 850 psi in the "run" mode. Attempts to restore the cleanup system were unsuccessful; the water level increased and overflowed into the main steam lines. The relief valves operated automatically at about 13 minutes after the scram and then repeated several operations for the next thirty minutes during which time reactor vessel water level was brought under control. After water level was brought under control, the emergency condenser was placed in service to assist in holding pressure.

No radiological release to the environment was reported. No damage to plant equipment was reported. The reactor was brought to criticality 2 days 14 hours later. The corrective actions made were as follows: (1) The time delay of the under frequency relay has been changed from 120 to 70 milliseconds, (2) the feedwater lockup circuits were rewired so that all circuits are not powered from the same M/G set, and (3) surveillance of the M/G sets was increased to afford early detection of control circuit problems.

2.7 Monticello Unit 1, February 26, 1972

Summary Description of the Incident at Monticello Unit 1 February 26, 1972

On February 26, 1972 an incident occurred at the Monticello Unit 1 nuclear plant in which a delay in opening of the steam bypass valves resulted in the opening of relief valves thereby releasing primary coolant water to the pressure suppression pool (torus). Background information concerning this plant is as follows:

Background Information

Facility:	Monticello
Type and Rating:	545 MWe - BWR

Location:	Monticello, Minnesota, 30 miles NW of Minneapolis Minnesota
Utility:	Northern States Power Company
NSSS and Contract Type:	General Electric Co. - Turnkey
Architect-Engineer:	Bechtel Corporation
Date of Operating License:	September 8, 1970

At approximately 5:30 a.m. on February 26, 1972, the turbine stop valves were undergoing routine testing with the reactor operating at 100 percent power. During this testing all of the turbine stop valves inadvertently closed because of a momentary decrease in the turbine emergency oil pressure. An immediate reactor scram resulted from the closure of the turbine stop valves as designed. The steam bypass valves to the main condenser failed to open for approximately 2-1/2 minutes because a turbine trip signal was not obtained. The delay in the opening of the bypass valves resulted in a rapid pressure increase in the primary system because of the loss of the main condenser as a heat sink. Within approximately 1/2 second the reactor pressure had increased from 1007 psig to 1062 psig and continued to increase to a maximum of 1118 psig. During this pressure increase the relief valves opened automatically, as designed. The relieving setpoints of the relief valves range from 1070 to 1090 psig. Indications were received that the #1 safety valve (setpoint 1220 psig) also opened for a short period (2-3 seconds). Drywell pressure increased to a maximum of 0.9 psig as a result of the steam released to the drywell; drywell pressure decreased to its normal value within 9 minutes. During the transient reactor vessel water level fluctuated over a wide range, but did not reach the elevation of the main steam lines and therefore did not affect the operation of the High Pressure Coolant Injection (HPCI) system.

The reactor resumed operation on the evening of the following day (February 27). On May 13, 1972 during a reactor shutdown for operator training, the drywell was de-inerted and entered. Minor damage to thermal insulation was observed in the area of the safety valve discharge. No significant radioactive releases to environment resulted from this event.

The reactor was shut down on June 5, 1972 and the safety valve that had prematurely actuated was replaced with a spare valve.

2.8 Dresden Unit 3, May 4, 1972

On May 4, 1972, an incident occurred at the Dresden 3 nuclear plant in which three safety valves opened prematurely, thereby releasing primary coolant water to the containment. Background information concerning the plant is as follows:

Background Information

Name of Facility:	Dresden Unit No. 3
Size and Type:	800 MWe - BWR
Location:	Morris, Illinois
Owner:	Commonwealth Edison Company
NSSS Contractor and Type:	General Electric Company - Turnkey
Architect-Engineer:	Sargent and Lundy
License Issued:	January 12, 1971

On May 4, 1972, at 9:01 a.m., while the reactor was operating at 100% of rated power, steady state condition, for reasons as yet undetermined, a reactor scram was caused by a spurious low reactor vessel water level signal. Following the scram, there were fluctuations in reactor vessel water level and pressure. The reactor water level decreased from a normal height of 14-1/2 feet above the active core to 12 feet and then increased to a maximum of 16-1/2 feet (still about 4 feet below the main steam lines). Reactor pressure decreased from 1000 psig to 815 psig and then increased to a maximum of 1110 psig.

The decrease in pressure from 1000 to 815 psig was because of continued steam and feedwater flow. The main steam isolation valves closed in accordance with design and reactor pressure started to increase. An attempt was made to place the isolation condenser in service but its discharge valve failed to open. Reactor pressure continued to increase until, approximately 13 minutes after the scram, one electromatic relief valve actuated automatically to control primary system pressure (setpoint - 1130 psig). Immediately following the operation of the electromatic relief valve, the high drywell pressure alarm, which is set to actuate at 2 psig, was actuated. The emergency core cooling systems started automatically in accordance with design;

however, injection of emergency core cooling water did not occur because the reactor pressure and the reactor vessel water level were above the setpoints required to cause injection into the reactor vessel. In addition, containment isolation occurred in accordance with design when the high drywell pressure alarm was received. The drywell pressure reached 2.5 psig and the maximum drywell temperature was 185°F. The pressure returned to normal within 30 minutes. Containment integrity was maintained for 12 hours until the containment atmosphere was sampled and vented prior to personnel entry.

Visual inspection within the drywell approximately 12 hours after the scram revealed that at least one safety valve had actuated during the occurrence, thereby releasing primary steam to the drywell (set point 1210 - 1240 psig). It is not known why the safety valve actuated at a reactor pressure of at least 100 psi below the valve setpoint. In addition, the rupture diaphragms on three other safety valves had broken indicating that unseating of these valves had occurred. The effect of the steam released was limited to minor insulation damage on valves in the immediate area of the safety valve that had actuated. No significant radioactivity releases or personnel exposures resulted from this incident.

In addition to the rupture of three safety valve diaphragms, the only other damage resulting from the incident was minor insulation damage. Instrumentation, structures and systems in the primary containment did not appear to be affected by the release. A downtime of four days resulted from the incident.

The corrective actions taken were as follows: Repair and testing of the isolation condenser valve; replacement of safety valve diaphragms; Replacement of the safety valve that lifted; and testing of the relief valve that operated.

3.0 Evaluation of Incidents

The incidents and unusual occurrences described in Section 2.0 were analyzed as a group to assist in identifying those elements and factors of importance to safety common to more than a single incident. A summary of some of the significant features of the incidents is presented in Table II. The evaluation of those elements of the incidents related to the safety of nuclear plant operations are discussed in this section and together with supplemental information presented in Appendix C, form the basis for the conclusions presented in Section 4.0.

In carrying out the evaluation, independent calculations were not performed to verify data and information presented in available documentation. The depth of the review of the three incidents at Dresden station was greater than those at other facilities and therefore had considerable influence on the findings and conclusions of the study group.

3.1 Initiating Mechanism

Two of the incidents considered were initiated by malfunction of the same kind of equipment. Malfunction of the turbine pressure control unit initiated the reactor system transient that subsequently led to the incidents at both Dresden 2 (June 5, 1970) and Millstone 1 (October 10, 1971). Although in both instances, spurious signals originating in the turbine pressure control unit resulted in reactor scram and turbine trip, the malfunction of these control devices involved two different components within the pressure control unit. In the case of Dresden 2, the abnormal behavior of the pressure control unit was traced to a faulty wire-wound potentiometer, whereas in the case of Millstone 1, the instability of this unit was attributable to the loosening of the mechanical linkage within the control unit.

The initiating mechanism for each of the other incidents reviewed occurred in different systems or components and were generally related either to spurious instrument and electrical signals or improper adjustment of safety setpoints.

It has been concluded that the incidents were not initiated by a common system or component malfunction. It is the opinion of the study group that malfunctions of the type that initiated these incidents are to be expected occassionally in any industrial installation composed of complex process systems and electro-mechanical components. Consequently, malfunctions leading to transients in reactor water level and pressure should be anticipated in nuclear plants.

TABLE II

SUMMARY OF INCIDENT CHARACTERISTICS

Parameter or Event	Dresden 2 6/5/70	Monticello 9/28/71	Millstone 10/10/71	Nine Mile 12/31/71	Dresden 3 12/8/71	Nine Mile 2/28/72	Monticello 2/26/72	Dresden 3 5/4/72
Reactor Pressure-Pmax/Pmin (psig)	1100/775	1040/670	1040/264	1117/NR	1040/795	1100/600	1118/910	1110/828
Drywell Pressure Pmax (psig)	20 (est.)	NA	NA	NA	20	NA	0.9	2.5
Drywell Temp.-Tmax (°F)	320 (est.)	NA	150 (est.)	150 (est.)	295	150 (est.)	150 (est.)	185
Reactor Vessel Water Level-Lmax/Lmin(in)[1/]	+130/NR	NR	+60/-9	+130/NR	+130/-20	+130/NR	+60/+9	+53/+1
Time to AEC Notification (days)	<1	∿43	<1	3	<1	9	∿79	<1
Downtime - (days)	52	<1	13	<1	20	2 1/2	1 1/2	4
Safety Valve Lifted Below Setpoint	Yes	No	No	No	Yes	No	Yes	Yes
Safety Valve Failed to Reseat	Yes	No	No	No	Yes	No	No	No
Relief Valve Lifted Below Setpoint	No	No	Yes	No	No	No	No	Yes
Relief Valve Failed to Reseat	No	Yes	Yes	No	No	No	No	No
Main Steam Lines Flooded	Yes	No	No	Yes	Yes	Yes	No	No
MSIV Closed Before Safety Valve Lifted	Yes	NA	NA	NA	Yes	NA	No	Yes
Isolation Condenser Functioned	No	NA	Yes	Yes	Yes	Yes	NA	No
Violation of Operating Procedures	Yes	No	No	Yes	Yes	No	Yes	No
Grounds on 125 V.d.c. System	Yes	No	No	No	Yes	No	No	No
ECCS Functioned as Required	Yes	Yes	Yes	Yes	Yes	Yes	Yes	Yes
Core Exposed	No	No	No	No	No	No	No	No

1/ Reading on the Level Recorder; a reading "o" inches on the recorder represents a water level of approximately 143 inches above the top of the active core.

2/ HPCI was out of order.

LEGEND:
NR - Not Reported
NA - Not Applicable

3.2 Performance of Plant Systems

The performance of the following key systems was evaluated to determine system response for each of the incidents reviewed.

3.2.1 Feedwater Control System

In all of the eight incidents, there were wide fluctuations in reactor vessel water level following reactor scram and turbine trip. Although water level transients are an inherent characteristic of Boiling Water Reactors, the magnitude of these fluctuations was amplified during the incidents by the actual performance of the Feedwater System.

During four of the incidents, the reactor vessel water level rose four to five feet above its normal level, sufficient to overflow into the main steam lines.

During two of the Dresden incidents, the feedwater control system was switched from automatic to manual control. During the third Dresden incident, timely response by the control room operators prevented excessively high water levels.

In the Dresden 3 (December 8, 1971) and Nine Mile Point (February 28, 1972) incidents, reactor vessel water level was affected by the interruption of feedwater pumping action because of an inadequate net positive suction head (NPSH) and because of the loss of electrical power, respectively. During all other incidents, the feedwater system required prompt manual control by the operator to prevent steamline flooding. The actions by the operator to take manual control of the feedwater system reflects the inability of the control system to maintain water level within the intended operating limits automatically. No specific design objectives or performance limits for maintaining water level are specified in Safety Analysis Reports; however, General Electric Company, the nuclear system designer, has stated that "there is no intention that the water level reach the main steam lines during the early course of a transient."*/

Flooding of the steam lines, of itself, does not necessarily present an unsafe condition. The influence of the presence of water in the main steam lines on the premature opening of either the relief valves or the safety valves has not been established. The ability of the High Pressure Coolant Injection System (HPCI) to fulfill its intended function with the turbine inlet line filled with water has not been demonstrated in the unlikely event that its operation is required under these circumstances.

*/ Letter, A. P. Bray to T. R. Wilson, June 13, 1972

It is concluded therefore that the performance of the feedwater control system to control reactor vessel water level automatically within acceptable bounds during anticipated transients has been less than satisfactory.

3.2.2 Overpressure Protection System

A. Safety Valves

In four of the incidents reviewed, primary coolant was released directly to the containment (drywell) atmosphere by the actuation of one or more safety valves. To date, a conclusion has not been reached by the owners of these facilities as to the exact cause of the safety valves operating prematurely. Selected aspects of all these incidents are included below for comparative purposes:

1. Relieving Pressure - Based on the maximum recorded system pressure during the transients, the safety valve(s) at the Dresden 2 (June 5, 1970 incident), the Dresden 3 (December 8, 1971 and May 4, 1972 incidents), the Monticello (February 26, 1972) operated from 70 to 250 psi below the nameplate setpoint. There have been no reported cases of safety valves opening above their setpoints. To date, there have been no supportable conclusions reached by the licensees as to the cause for premature actuation of safety valves.

2. Reset Pressure - With the exception of the Dresden 3 incident of December 8, 1971, safety valve openings were momentary in nature (openings of a few seconds duration) for all events reviewed. During the Dresden 3 incident of December 1971, the safety valve remained open for 1 1/2 minutes and relieved system pressure from 1020 to 910 psig before resetting. During the Dresden 2 incident of June 5, 1970 the safety valve that initially opened apparently reset within a few seconds, but the steam jet damaged the lifting levers of two other safety valves in the vicinity causing them to remain partially unseated. Based on information obtained from Commonwealth Edison Company personnel (meeting with Study Group on May 26, 1972)

there have been several other instances where they have observed torn or broken rupture discs in the discharge ports of safety valves indicating "burping" of the safety valves. The cause for this is unknown, but is postulated to be associated with controlled flooding of the main steam lines during normal shutdowns.

For the incidents reviewed, all of the safety valves that experienced premature actuation were manufactured by Dresser Industries and were factory set with steam before shipping to the reactor site for installation. The extent of site maintenance activities on safety valves is not known. The study group was informed by representatives of Dresser Industries that the safety valve that actuated during the Dresden-3 incident of May 4, 1972 was found to have an improper internal adjustment (blowdown ring) when inspected at the factory after its return. The adjustment would have permitted the safety valve, if opened, to remain open until pressure had decreased at least 200-250 psi below the nameplate setpoint.

All of the plants that have experienced premature safety valve actuation have safety valves and relief valves mounted on the main steam lines within a few feet (2-1/2 to 3) of each other. In four of the eight incidents reviewed, it appears that the premature opening of safety valves was nearly simultaneous with the operation (opening or closing) of a relief valve on the same steam line. The main steam isolation valves were closed at the time of safety valve operation.

During two of the eight incidents the steam lines began to flood from 2-1/2 minutes to 5 minutes before a safety valve operated. In two other incidents, the safety valve opened within 3 seconds following a turbine trip.

Post-incident testing of the relieving setpoints for the safety valves that opened prematurely disclosed that the valves relieved from 50 to 110 psig below the desired setpoint.

In summary there does not appear to be a common cause or explanation for premature safety valve operation. Postulated mechanisms are: mechanical forces generated by the closing of relief valves, pressure pulses generated by the closing of main steam isolation or relief valves,

hydraulic forces resulting from impingement of water on the valve seats and improper setting and testing of the valves. In addition, the amount of coolant discharged during one of the events may have been related to site maintenance activities and handling of the safety valve.

B. Relief Valves

An increase in primary system pressure above normal requires either the manual operation or automatic actuation of the relief valves during six of the incidents. These valves are designed to preclude the need for operation of the safety valves during operational transients and are normally set to operate automatically at a pressure of approximately 100 psig below the safety valve setpoints. The safety valve setpoint pressure (1210-1240 psig) was not reached, based on recorded information, at any time during these incidents and it is significant to note that the overpressure protection devices (relief and safety valves) fulfilled their intended function of preventing overpressurization of the primary coolant system.

Premature actuation of relief valves at a pressure less than 27 psi below the desired setpoint was observed during two incidents. None of the events that were reviewed involved the operation of relief valves above the setpoint pressure. During two of the incidents (Monticello - September 28, 1971 and Millstone 1 - October 10, 1971), the relief valves failed to close in accordance with design, thus discharging a larger quantity of coolant to the pressure suppression pool (torus) than expected.

The operation of relief valves appears to be in accordance with plant design during operational transients. The frequency of operation appears to fall within the design specification of the valves (Target Rock Valves are designed for 20 cycles of operation per year for the 40 year life of the plant).

The release of coolant to the pressure suppression pool through relief valves has resulted previously in damage to torus baffles and to the torus liner. Licensees have been notified of these potential problems and corrective actions have been required of licensees in some cases by the Regulatory Staff. Further consideration of the stresses placed on the relief valves discharge piping during operation of these values is in progress.

3.2.3 Containment System

Pressurization of the containment (drywell) above 2.5 psig occurred in only two of the Dresden incidents (June 5, 1970 and December 8, 1971). In the Dresden incident of May 4, 1972, the maximum pressure recorded was 2 1/2 psig. During other incidents, pressures of 0.9 psig or less were measured.

In the two Dresden incidents, the maximum pressure of 20 psig observed in the containment (drywell) agrees with the calculated pressure that would result if the non-condensible gases in the drywell were driven into the torus. It thus appears that, in these incidents, the pressure suppression system functioned in accordance with design.

Analysis of the containment response during the two Dresden incidents indicated that the maximum temperature reached was about 320°F. Although the 320°F temperature reached was greater than the design temperature (280°F) stated in the FSAR, analysis by the licensee has shown adequately that a safety factor of about two exists at this temperature to preclude buckling of the containment liner. Visual examination of the liner following the incident confirmed that buckling did not occur.

The release of coolant to the pressure suppression pool (torus) through the relief valves could have safety significance in at least two respects as follows: 1) by release radioactive coolant to the pool, thus, potentially contributing to the long term buildup of radioactivity and posing cleanup and maintenance problems, and 2) by damage to structures. Analysis has shown that the baffles installed in the torus of some plants become overstressed and are subject to failure during relief valve operation.

The potential problems associated with buildup of long half-life nuclides in the pressure suppression pool resulting from frequent use of relief valves were not analyzed in this study. Licensees have been notified of the potential for failure of torus baffles and corrective actions are being followed by the regulatory staff.

A number of malfunctions of equipment and materials located within the drywell have been reported following the release of coolant into the drywell as follows: potting compounds used to seal containment electrical penetrations deteriorated permitting leakage, pilot (solenoid) valve operators for main steam isolation and automatic depressurization valves failed to function because of electrical shorts or grounds, and the flexible diaphragms on isolation valves deteriorated permitting leakage. Failures of electrical and instrument cables and motors have also been reported. These electrical items are discussed separately in Section 3.2.5 below. With a few exceptions, the failure of equipment and materials in

the drywell occurred during or following the Dresden incidents in which the drywell pressure and temperature reached 20 psig and 320°F, respectively. Although tests have been performed by licensees, vendors and equipment suppliers to demonstrate that the materials and equipment are capable of performing their intended function in the environment (temperature, pressure and humidity) associated with a major loss of coolant accident, because of such things as aging, radiation effects and cycling in the adverse environment, it appears that the potential for failure may be significantly increased.

It is concluded that for operating plants, the capability of safety equipment and materials to withstand frequent coolant releases during the life of the plant warrants further review. For plants not yet in operation, review of the equipment environmental specifications, environmental test programs, and equipment capabilities by the applicant and the regulatory staff should be rigorous and complete.

3.2.4 Reactivity Control System

In all of the incidents considered, the reactivity control system performed properly. In every incident, the shutdown status of the reactor was verified immediately. In cases where nuclear instrumentation failed, the failures did not occur until long after (several minutes) the reactor was scrammed and shutdown had been confirmed.

3.2.5 Electrical Systems

Problems that were noted to have occurred in plant electrical systems during the incidents included several electrical grounds and short circuits that developed in the 125 volt d.c. electrical system in the drywell. These abnormalities resulted in a loss of capability of control room annunciator panels to indicate properly the status of ECCS systems and radiation instruments in the plant during one of the incidents.

The several shorts and grounds experienced in this system raises a question as to the reliability of this power supply to function during a loss of coolant accident, particularly since both safety related and non-safety related items are powered from a common power supply. The 125 volt d.c. system in the drywell powers the solenoid pilot valves for the Automatic Depressurization System and therefore the ability of this system to function without failure is questionable. The 125 volt d.c. power supply system within containment, including conduits, cables, junction boxes and devices, should be designed to preclude failure from the release of primary coolant to the drywell.

Grounds were also experienced in several electrical motors, including those that actuate the following valves, which are required to perform a safety function:

a) 'B' Recirculation discharge valve
b) 'A' Recirculation tie line valve
c) HPCI turbine steam supply line valve.

From the information reported on the resistance to ground, it did not appear that the proper operation of the valves would have been affected. However, no firm conclusions could be reached because the available information concerning these tests was incomplete.

3.2.6 Emergency Core Cooling Systems (ECCS)

Automatic starting of the ECCS was required during the following incidents:

Dresden 2, June 5, 1970	-	High	drywell	pressure	(20 psig)
Dresden 3, December 8, 1971	-	"	"	"	(20 psig)
Dresden 3, May 4, 1972	-	"	"	"	(2-1/2 psig)

Pumps in the Low Pressure Coolant Injection (LPCI) and Core Spray subsystems started satisfactorily during all incidents, but were not required to inject coolant water because system pressure and reactor vessel water inventory (level) were maintained adequately.

The High Pressure Coolant Injection (HPCI) subsystem received an initiating signal during each of the 3 incidents, but would not have been capable of injecting cooling water during the Dresden 2 (June 5, 1970) and Dresden 3 (December 8, 1971) incidents because of flooding of the HPCI steam supply line.

For the Dresden 3 (May 4, 1972) incident the HPCI turbine tripped because of high reactor vessel water level, but the HPCI steam supply line did not flood (4" of margin).

The Automatic Depressurization System (ADS) was not called on to function during any of the Dresden incidents. The only problem noted with the ADS system occurred during the Dresden 3 (December 8, 1971) event, when one of the 5 installed relief valves was disabled as a result of steam impingement on the pilot operator from the safety valve that operated.

In summary, the Emergency Core Cooling Systems performed as designed. The injection of water by the HPCI system was not required during any of the incidents, but it is important to note that the HPCI turbine may have been damaged and unavailable for coolant injection during 2 of the incidents, if the high reactor water level turbine trip condition had cleared and the HPCI had automatically started on demand with the HPCI steam line flooded.

It is concluded that proper operation of the feedwater system and control of large overshoots in reactor vessel water level is important to assure proper HPCI system performance.

3.2.7 Isolation Condenser

With the exception of the three Dresden incidents, the isolation condenser system either functioned properly or was not required to be actuated.

During two of the Dresden incidents the isolation condenser inlet line was flooded, thereby greatly reducing the effectiveness of the condenser system. Even if the line had not been flooded, there is some question as to whether the system would have functioned in the first incident, since the condensate return line high flow trip was improperly set.

During the Dresden incident of May 4, 1972, the condensate return valve failed to open remotely and had to be manually opened. The system then functioned normally.

The isolation condenser system is designed for shutdown cooling to serve as a heat sink for fission product decay heat. If the system becomes inoperable due to steamline flooding, other means would be available (Automatic Depressurization System, main condenser) to control reactor system pressure and to dissipate the heat. It is concluded that the loss of this system does not compromise the safety of the facility.

3.2.8 Coolant Cleanup System

During the incidents reviewed, the reactor vessel water level transients experienced following scram were sufficient to cause isolation of the coolant cleanup system automatically in accordance with design. This action removed the only available mechanism for reduction of the water level in the reactor vessel once it was in an isolated condition. This circumstance is not cause for concern. However, means for coping with

excessive leakage through feedwater control valves into the reactor vessel and continued addition of water to the reactor vessel from the Control Rod Drive system as noted during the Dresden 2 June 5, 1970 incident should be considered.

During the Dresden 3 incident (December 8, 1971), the release of coolant water from the primary system to the main condenser occurred through a safety valve in the cleanup system approximately 23 minutes after the transient began, when the cleanup system was being placed back in service. This discharge of coolant was terminated at 950 psig when the cleanup system isolated automatically.

The coolant cleanup system appears to have isolated and performed as designed in all cases.

3.3 Control Room Operations

The response of control room operating personnel to incidents and unusual occurrences was evaluated for the Dresden incidents. The study group reviewed the number of operating personnel available in the control room during incidents, the experience and training of these personnel, and deviations from operating procedures specified in the technical specifications. The deviations from operating procedures were reviewed for all incidents considered.

In the course of reviewing and evaluating control room operations during incidents, the study group visited the Dresden Simulator Facility and observed a simulation of the Dresden incidents of June 5, 1970, and December 31, 1971, as they appeared in the control room. Operating procedures relating to the incidents were also discussed with representatives of the reactor vendor and the licensee.

The number of operating personnel present in or immediately available to the control room at the time the incidents occurred was not a factor in determining subsequent events, since in each instance more than one qualified reactor operator was present. In discussions and correspondence with representatives of the General Electric Company, the control panel designer, regarding the number of operating personnel required to control anticipated transients and incidents, they recommended that "a shift supervisor onsite and two qualified reactor operators in the main control room are considered adequate to monitor information and take corrective action required during all anticipated transients and accident conditions to shut the plant down safely."*/ The technical specifications covering control room staffing for some operating plants require only a single licensed operator in the control room and are therefore in conflict with General Electric's recommendation.

*/ Letter, A. P. Bray to T. R. Wilson, June 13, 1972

The training and experience of the reactor operators appeared to be adequate and meets AEC guides and standards. Subsequent to the incidents at Dresden Station, Commonwealth Edison Company has instituted an extensive retraining program for all operating personnel, and revised and updated the emergency and operating procedures. In addition, supervisory operating personnel have reviewed the primary coolant water release incidents with the operating personnel, highlighting areas of concern where operator response was considered marginal.

The effectiveness of the retraining program and the updated operating procedures was tested to a degree during the May 4, 1972 incident at Dresden Unit 3. During this incident, the effectiveness and adequacy of operator actions during and following the incident showed a significant improvement over the operator actions during the two previous incidents. It appears that the operating personnel at the Dresden Units 2 and 3 power plants would now be expected to respond to a similar primary coolant release incident in a manner which would significantly reduce the magnitude of the incident. Other licensee operators such as those at Millstone and Monticello, also appear to have benefited from the experience and evaluation of these incidents. For example, a review of the Millstone Unit 1 blowdown incident that occurred on October 10, 1971, demonstrated good operator response.

During the incidents a number of deviations from operating procedures and technical specifications requirements were experienced. These deviations are discussed further in Appendix C for each of the Dresden incidents. Examples of operating deviations are as follows: venting of the containment prior to determining the radioactivity level and composition, removing the drywell pressure sensors from service with the pressure above 2 psig, failure to operate containment sprays when the pressure was above 2 psig, shutdown of the core spray and low pressure coolant injection pumps prior to determining the cause of the incident, operation of the Standby Gas Treatment System (SGTS) when the drywell pressure was above the pressure for which the SGTS was designed, failure to shut down the feedwater pumps when the reactor vessel water level exceeded 60 inches and failure to reset the feedwater valve lockout.

Some actions of operating personnel to control the reactor during unusual occurrences and transients tended to aggravate and prolong the incidents. The causes of these operating deviations could not be determined definitely. Possible causes include insufficient training, inadequate procedures, control panel and control room design characteristics and the complexity of the task

confronting the reactor operator. Reactor operating procedures are audited by AEC Field Inspectors and review as required by licensing examiners in the course of examining applicants for reactor or senior reactor operating licenses. In several instances, licensees have made extensive revisions to operating procedures following the incidents and operator retraining has been instituted. It would be nearly impossible for the Regulatory Staff to review all changes to operating procedures in a timely fashion, since such changes are normal and frequent. The procedures are reviewed only in a general way and in addition, the licensees' organization for generating and approving procedures is reviewed.

3.4 Control Panel Instrumentation and Controls

In evaluating the actions of operations personnel during the incidents, consideration was given by the study group to the control panel layout and the type and method of displaying information to the operator. Several process variables of importance to diagnosing the course of a transient, making operating decisions and taking prompt corrective actions during transients may not be adequately displayed on the control panel. For example, a means for the operator to determine whether relief and safety valves have opened is not readily available. Temperatures measured by detectors installed in the safety valve discharge lines currently are recorded on a multi-point recorder having a cycling time of about 5 minutes. Reactor water level is indicated on several instruments, however, the rate of change of reactor vessel water level is difficult to determine without the undivided attention of the operator for a period of several minutes. Other parameters that should be considered for modification with respect to the method of display include: drywell pressure, drywell temperature, reactor vessel water level. Means for quickly detecting safety valve actuation should also be considered.

It is concluded that during the control panel design, further consideration should be given to the instrumentation and controls provided and the layout of the control room and the control panel taking into consideration such things as the number of operating personnel available for controlling transients and unusual occurrences, the information required to rapidly diagnose and take proper corrective action in response to unusual occurrences, and other human engineering aspects of the plant control system design.

A standard or guide developed by the nuclear industry for control room and control panel design that addresses the human engineering aspects appears to be needed.

3.5 Plant Damage - Effects of Coolant Releases

The possible vulnerability of equipment within the containment was demonstrated by the Dresden incidents of June 5, 1970 and December 8, 1971, which resulted in substantial increases in containment pressure (20 psig), temperature (320°F) and humidity. During all other incidents reviewed, these increases were moderate.

In all incidents, there was damage to thermal insulation covering pipes and valves caused by steam impingement, however, this did not lead to any loss of function of equipment or components. Damage to drywell paint, ventilation ducts, an electromatic relief valve and a solenoid valve operator were caused by the steam jets emitting from the safety valves during the Dresden incidents of June 1970 and December 1971.

Damage of the same general type that occurred in the Dresden incidents of June 5, 1970 and December 8, 1971 caused by the environmental conditions that existed within the containments was the loss of substantial numbers of nuclear instrumentation cables. During the incident of June 5, 1970, 50 percent of the SRM and IRM cables and 63 percent of the LPRM cables were lost. Following this incident, the SRM and IRM cables were replaced with cables whose materials were rated for higher temperature, however, the LPRM cables were replaced with cables identical to those used originally. Consequently, the LPRM cables were affected again during the incident that occurred on December 8, 1971.

Moisture was the probable cause of the shorting or low resistance of valve operators, blower motors and cables during the June 5, 1970 incident and a fan motor during the December 8, 1971 incident. Grounding of several 125 volt d.c. cables was also experienced during the Dresden incident of June 1970. Cable damage and shorting of motors, which was common to the June 5, 1970 and December 8, 1971 incidents, did not preclude determination of the shutdown status of the reactor or result in severe operational degradation of any systems required for protective or emergency functions.

An item of damage not common to the other incidents found subsequent to the Dresden incident of June 5, 1970, was the leaking of seven drywell electrical penetrations from the center, inboard to the drywell. Leakage of several containment isolation valves equipped with Buna-N seals was also discovered at a later date, that may be attributable to degradation resulting from these incidents.

Based on experience gained from the three Dresden incidents, it can be concluded that damage can be expected by steam impingement on thermal insulation, paint (or coating) and structures such as ventilation ducts. Damage due to degradation of insulating materials, cables, electric motors and solenoid valves also can be expected if significant coolant release occurs. The potential impact of either repeated releases of small quantities of steam or the release of larger amounts of steam than occurred during the Dresden incidents of June 5, 1970 and December 8, 1971, on the ability of drywell components to fulfill their intended function was not assessed and warrants further consideration.

3.6 Corrective Action

After each unusual occurrence or incident at a licensed facility, the licensee is required to return the plant to the condition required by the license and technical specifications prior to placing the plant back in operation. With the exception of the incidents at Monticello, all components and systems required by technical specifications which malfunctioned during the incidents were repaired or replaced and all damage resulting from the incident was repaired prior to further operation. Following the Monticello incidents of September 28, 1971 and February 26, 1972, the relief valves were not replaced or repaired immediately for reasons discussed in Section 3.7.

Section 6.3.A. of the Technical Specifications requires that the Station Review Board prepare a report for each abnormal occurrence, that includes an evaluation of the cause of the occurrence and recommendations for appropriate corrective action to prevent or reduce the probability of a repetition of the occurrence. After each incident, evaluations and corrective actions were performed to reduce the probability of similar occurrences. These corrective actions are discussed in Appendix B for the three Dresden incidents. The corrective actions have reduced the magnitude of the feedwater upsets, but have not eliminated all the conditions which led to excessive water levels or steam discharges from safety valves. Programs are now in progress by both the General Electric Company and the Commonwealth Edison Company to improve the feedwater control system and to investigate premature actuation of safety valves. General Electric has undertaken, in conjunction with the licensee, a test program at Dresden 2 to determine the feedwater control valve response time and to test a new valve positioner. General Electric Company now plans to design circuits and equipment to further assist in the control of feedwater during transients involving scrams. Also, Commonwealth Edison Company has contracted with the Franklin Institute to study the Feedwater Control System and recommend design modifications.

With respect to the premature lifting of safety valves, the General Electric Company has initiated short and long term test programs to establish the cause for improper valve operation. The short-term program involves instrumenting safety valves in an operating plant or plants to determine the effect of the steam line flooding rate on lifting of safety valves. These data would be used as a basis for further revising operating procedures for the Feedwater Control System. Tests and examinations are also being performed at the valve manufacturer's facility on one of the safety valves which opened prematurely during the December 8, 1971 incident. The purpose of these tests is to disclose any contributing design, manufacturing or testing deficiencies.

The long-term program involves analysis and testing to establish the most probable causes of premature lifting of the safety valves.1/ The schedule for short and long term tests was not established as of the middle of June.

Commonwealth Edison Company has also initiated a program to establish causes of premature lifting of safety valves. This program involves accumulation of data at Quad-Cities during transient tests, instrumentation of a steam line at Dresden 2, and employing a consultant to investigate mechanical forces on the safety valves by analysis and by tests.2/

The objectives of the ongoing programs to investigate feedwater control system problems and safety valve lifting problems appear to be satisfactory, however, final evaluation of the results of these investigations and proposed corrective actions must await their completion. Firm schedules for completion of these programs were not available at the time this report was written.

3.7 Documentation

The documentation in the docket files related to each of the incidents was reviewed with respect to licensee notification of AEC, the sufficiency of licensee reports of incidents, and the availability of supporting documentation regarding actions taken by the licensee and the regulatory staff. A chronological listing of the documentation associated with each incident is provided in Appendix D.

1/Letter, A. P. Bray to T. R. Wilson, dated June 13, 1972.
2/Letter, H. K. Hoyt to T. R. Wilson, dated May 31, 1972.

The documentation associated with each incident was considered adequate if: (a) the licensee conformed with the reporting requirements of the technical specifications; (b) the licensee's reports presented the information discussed in Safety Guide 16, "Reporting of Operating Information," and specifically addressed the items listed in Section C.2.a.(1); and (c) the records were sufficient to enable an after-the-fact independent technical evaluation of the event and an independent appraisal of the timeliness and effectiveness of licensee and staff actions.

Four of the eight incidents considered were not reported to AEC as required by the technical specifications. These incidents were the Nine Mile Point incidents of December 31, 1971 and February 28, 1972, and the Monticello incidents of September 28, 1971, and February 26, 1972. In each of these instances, the licensee did not consider the events to be of sufficient significance as to be defined as an incident or unusual occurrence. Better definition in the technical specifications of incidents and unusual occurrences which are required to be reported by licensees appears warranted.

The records available in the docket files were not sufficient to allow the study group to perform a complete and independent technical evaluation of the incidents. The study group is of the opinion that the information to be reported on incidents should be supplemented in a number of areas, including information on such things as operating procedures, operator training, maintenance practices, system and component performance, radioactivity releases and long-term plans and schedules.

The policies and procedures for the review of incidents and unusual occurrences by the AEC Regulatory Staff are not clearly defined and completely documented. In each case, the Regulatory Staff made an evaluation and an explicit decision that it was safe to resume operation of the plant, however, the bases for these decisions were not documented completely in all cases. Staff documentation of follow-up investigations and corrective actions at nuclear plants other than the one involved in the incident also was not sufficient of itself to permit evaluation of the scope and effectiveness of the licensee actions. Lack of complete documentation is not intended to imply inadequate review, erroneous conclusions or unsafe operation of nuclear plants. It is discussed herein to assure an understanding of existing procedures by licensees and the Regulatory Staff and as a basis for recommending clarification of the procedures and documentation.

3.8 Safety Significance

Evaluation of the safety significance associated with the incidents of primary coolant release was based principally on the consideration of two questions: Did the incidents endanger the health and safety of the public? What was the significance of the deficiencies that were observed during the incidents with respect to the design, construction, testing or operation of nuclear plants? In considering the second question, no attempt was made to postulate additional equipment failures or human errors and to assess their consequences, but rather an attempt was made to identify the significance of problems encountered and to recommend areas warranting further safety consideration.

3.8.1 Frequency of Coolant Release Incidents

There are currently eight operating Boiling Water Reactors that may be classified as second generation plants. The first of these, Oyster Creek, began commercial operation in mid-1969. The most recent plant to near commercial operation is Quad-Cities 2. These eight plants have accumulated a total of approximately 15 reactor years of operating experience (including downtime).

Since startup of Oyster Creek in mid-1969, there have been eight incidents in which primary coolant has been inadvertently released because of either premature actuation of relief and safety valves or failure of the relief and safety valves to close once they were opened. During these eight incidents, there have been four instances of premature safety valve actuation, two instances of premature relief valve operation, and four instances of failure of relief or safety valves to close as intended.

These limited operating statistics indicate that on the average approximately one coolant release incident attributable to relief or safety valve malfunction can be expected for each two reactor-years of operation. Comparison of the observed frequency rate for coolant release incidents with the predicted frequency rate was considered outside the scope of this evaluation because of the unavailability of information on the predicted frequency of such events.

The significance to safety is related both to the frequency that the primary coolant system pressure boundary has been breached and to the added burden placed on the reactor containment to prevent the release of radioactive materials. This forms the basis for the conclusions that further consideration should be given (1) to

identifying the basic cause of relief and safety valve malfunctions and to developing plans to prevent or minimize the number of occurrences of coolant release, and (2) assuring that the containment and the safety equipment located therein are capable of performing their intended functions during and following coolant release incidents throughout the life of the plant.

3.8.2 Release of Radioactive Materials

The significance of the coolant release incidents considered in the evaluation with respect to public health and safety was negligible because the quantity of the radioactive effluents released was significantly less than that permitted by the plant technical specifications during normal operation. Therefore, the concentration levels at the site boundary and the radiation dose to which operating personnel were exposed were well below 10 CFR Part 20 limits, especially when the resulting doses are combined with averaged annual doses. Supporting information and data is presented in Appendix D.

3.8.3 Plant Performance

The integral performance of nuclear plants that have experienced incidents of coolant release has been satisfactory and in accordance with the design intent as demonstrated by the fact that no significant release of radioactive effluents has occurred that has endangered the health and safety of the public. Nevertheless, a number of equipment malfunctions and operating errors have been experienced during incidents of coolant release. Individually, each of these equipment failures and human errors has little or no significance to the overall safety performance of the plant because of the multiple safety features provided in nuclear plants and the redundancy of the safety systems and components. Collectively, the problems experienced during the incidents of coolant release may have safety significance, since areas have been revealed where improvements in the design, fabrication or operation of the plants could be made that would increase the safety margin available.

The performance of the Feedwater Control System (FCS) to control water level automatically within the intended design range during anticipated transients has been less than satisfactory. Flooding of the main steamlines and in some instances, flooding of the line to the High Pressure Coolant Injection (HPCI) system turbine may have contributed to premature safety valve actuation and prevented operation of the HPCI system had it been called on to function. Although the performance of the FCS is of no direct safety significance, the added

burden placed on the reactor operator by virtue of its instability coupled with the potential affect on safety valve actuation and the HPCI system operability, suggests that by improving the performance of feedwater control system a substantial improvement in safety may be achieved.

The release of coolant to the pressure suppression pool through relief valves has resulted in structural damage to wave dampening baffles installed in the pools of a few nuclear facilities. Since corrective action has been initiated to eliminate this problem, the release of primary coolant through relief valves does not appear to be of further safety significance.

The failure of relief valves to open when called upon, although of little significance with respect to overpressurization of the primary coolant system, is of safety significance to core cooling during small line breaks in the primary system. Although the HPCI system is intended to provide core cooling in the event of the rupture of a small line, use of the relief valves is necessary to reduce the pressure in the primary coolant system sufficient to permit the backup core cooling systems (LPCI and core spray) to function. The margin of safety available thus will be improved by assuring a high degree of reliability of the relief valves.

The release of primary coolant to the containment (drywell) through safety valves was never intended except in rare emergencies to prevent primary system overpressurization. The frequent primary coolant discharges that have occurred have demonstrated the increased potential for containment leakage and for the malfunction of safety related components located within the drywell both of which are of significance to safety. This forms the basis for the conclusions that the release of primary coolant to the containment should be prevented or that the performance of safety equipment to withstand satisfactorily the occasional pressure, temperature and humidity conditions associated with coolant releases should be adequately demonstrated.

The cause for the operating errors and deviations from operating procedures experienced during the incidents reviewed are not well understood. Although the consequences of these errors and deviations were not significant for these incidents, collectively they are believed to reflect shortcomings in one or more of factors such as the control room design, operating procedures and personnel training. Therefore, these factors warrant additional consideration and study.

In summary, considered collectively the equipment malfunctions and human errors that have been experienced could potentially be of safety significance and warrant further attention to further reduce the probability of occurrence of a series of equipment and human failures that could affect the health and safety of the public.

4.0 Conclusions and Recommendations

In arriving at the conclusions and recommendations, the study group has relied primarily on those members most familiar with the subject. Because members of the group represent various disciplines, not all members contributed to all areas covered in this report.

The study group found it desirable to consider the entire sequence of events that occurred during the incidents and the integrated performance of their individual importance to safety.

The evaluation by the study group is presented as a set of conclusions followed by recommendations. These conclusions are specific in some instances and necessarily broad in others. The conclusions have been placed in three general categories dependent upon their relation to plant design and equipment performance, reactor operations and management and administration. Supporting discussions and information may be found in the Appendices B, C and E.

Conclusion 1 - Public Health and Safety

The effect of the incidents of coolant release of the health and safety of plant operating personnel and the public was negligible. The radiation exposure of onsite personnel and the concentration levels of radioactive effluents released beyond the site boundary were well below the permissible limits contained in 10 CFR Part 20.

Conclusion 2 - Performance of the Feedwater Control Systems

(a) The performance of feedwater control systems to control automatically the reactor vessel water level within acceptable bounds during anticipated transients has been less than satisfactory. Manual operator control of the feedwater system during anticipated transients has not been successful on several occasions in preventing the flooding of main steam lines and the inlet lines to other systems of importance to safety (HPCI and isolation condenser).

(b) Specific numeric performance objectives have not heretofore been specified in Safety Analysis Reports or Technical Specifications for the feedwater control system with respect to the maximum and minimum reactor vessel water level to be maintained during anticipated transients.

(c) Tests conducted on the plants as part of the startup and power ascension test programs have not verified that the feedwater control system can control automatically reactor vessel water level within acceptable bounds during anticipated transients.

(d) The effectiveness of corrective actions taken by licensees to date cannot be assessed, since the results of studies undertaken and corrective actions taken have not yet been reported to AEC.

Recommendations

(a) The feedwater control system should be designed to control automatically the reactor vessel water level during anticipated transients without flooding of the main steam line or the lines to other safety related equipment.

(b) Licensees of operating Boiling Water Reactors (BWR's) and applicants applying for construction permits or operating licenses for new BWR plants should be required to specify and justify by analysis the performance objectives for the automatic control of reactor vessel water level (maximum and minimum water levels) during anticipated transients. Suggested performance objectives are as follows: (1) the maximum water level attained should not initiate isolation of any safety feature, such as the High Pressure Coolant Injection System (HPCI) or disable any system or component required for the orderly shutdown of the reactor, and (2) the minimum water level attained should not require the activation of any safety system. The analyses required to select more specific performance objectives for the control of reactor vessel water level are outside the scope of this study and have not been performed. Realistic acceptable bounds for the automatic control of reactor vessel water level should be developed by the Directorate of Licensing.

(c) Licensees of operating Boiling Water Reactors (BWR's) and applicants applying for operating licenses for new BWR facilities should be required to conduct tests during the startup and power ascension test program to verify that the feedwater control system is capable of

meeting the performance objectives specified under item (b) above for the automatic control of reactor vessel water level during anticipated transients. For example, the performance of the feedwater system may be verified by measuring its performance without operator action following a reactor scram and turbine trip from full power accompanied by main steam line isolation. Licensees and applicants should be required to submit test plans for evaluation by the regulatory staff.

(d) The regulatory staff should follow and require timely completion of the program now underway by Commonwealth Edison Company and General Electric Company to determine the cause of the marginal performance of the feedwater control system.

Conclusion 3 - Safety Valves

(a) The cause(s) of premature actuation of safety valves below their setpoint pressure has (have) not been determined.

(b) The release of reactor coolant through safety valves to the containment (drywell) was not intended by the nuclear system designers except during highly improbable incidents. Some evidence exists which indicates that the occassional release of coolant to the containment throughout the life of the plant may degrade the containment integrity and reduce the reliability of safety related systems and components located in the containment. Therefore, the release of coolant through safety valves should be either prevented or the frequency of coolant releases reduced to the extent practicable.

(c) The current methods of testing the actuating pressure of safety valves with media other than steam are subject to question.

(d) The training of nuclear plant technicians engaged in the adjustment, testing, handling, and maintenance of safety valves is less than satisfactory based on the evidence available.

Recommendation

(a) The regulatory staff should follow and require timely completion of studies and test programs now underway by Commonwealth Edison Company and General Electric Company aimed at determining the cause of premature safety valve actuation.

(b) Dependent upon the results of the studies and test programs on safety valves now in progress, the regulatory staff should require that incidents of coolant release in both operating BWR facilities and new BWR facilities under construction be prevented or their frequency minimized to the extent practicable. Once the

cause(s) of premature safety valve actuation is (are) identified, licensees and applicants should be required to submit corrective action plans for review by the regulatory staff.

(c) The regulatory staff should require that safety valves be tested to verify their actuation pressure using steam as the test medium prior to their installation and thereafter in accordance with existing requirements, unless the results of studies and test programs now in progress conclusively verify that the actuation pressure, using a media other than steam, can be correlated with the actuation pressure using steam under the actual external environmental conditions expected during plant operation. Capacity certification tests using saturated steam should continue to be the responsibility of the valve manufacturer in accordance with ASME code requirements.

(d) The regulatory staff should require that applicants for operating licenses submit in their applications definitive plans and schedules for the training of nuclear technicians and repairmen engaged in the testing and maintenance of safety related systems and components, such as safety valves, to assure compliance with the intent of ANSI 18.1 (1971) and Appendix B, 10 CFR 50.

Conclusion 4 - Control Room Operations

(a) The response of operating personnel to incidents and unusual occurrences aggravated and prolonged the events on several occasions. The specific cause(s) of the operator errors experienced could not be determined. Several potential contributing factors were considered in the evaluation as follows: insufficient training or retraining, inadequate operating procedures, control panel design, lack of information concerning the status of plant systems and components, and the complexity of and limited time available for operating decisions.

(b) Operator training and retraining using reactor simulators appear to have improved operator performance.

(c) Operating procedures for coping with anticipated transients are either incomplete or deficient at the time of plant startup. Improvements in the procedures have been and are continuing to be made in limited areas; however, further improvements are needed.

(d) The method of displaying information on the control panel and the location of controls in relation to each other do not appear to have been considered sufficiently in the design of the control room for coping with anticipated transients and postulated accidents, particularly when it is considered that there are no requirements for more than a single qualified operator in the control room.

Recommendation

(a) To the extent practicable, applicants and licensees should be required to use reactor simulators to train and evaluate reactor operators and to verify operating procedures for coping with anticipated transients. The reactor simulators should be capable of simulating in real time the plant response to anticipated transients and postulated accidents considered in the Safety Analysis Report. Operator response should be observed and evaluated by the AEC examiner during the reactor operator licensing process and the results used by the licensee as a basis for modifying operating procedures and evaluating the need for operator training or retraining. The results of simulation exercises should be recorded and audited periodically by AEC.

(b) Licensees of operating plants and applicants applying for permits or licenses for new facilities should be required to install instrumentation in the control room to continuously record the following parameters or component response: containment pressure, containment temperature, reactor vessel water level and safety valve actuation. The instrumentation should be capable of

continuously recording the above parameters over the entire range expected during anticipated transients and postulated accidents. The purpose of this instrumentation is to provide the reactor operator with information regarding the magnitude and rate of change of these parameters or component response essential to reaching proper operating decisions.

(c) A standard or guide should be developed by industry for the control panel and control room design that better addresses the human engineering aspects of operations during abnormal situations.

(d) A guide should be developed to assist in evaluating the number of reactor operators needed to cope with anticipated transients and postulated accidents. The criteria should give consideration to the following: (1) the assumptions used in the design of the control incidents and unusual occurrences, (2) the operating procedures for controlling incidents and unusual occurrences, (3) reactor operator training and retraining program plan, and (4) the human engineering aspects of the control room and control panel design. Based on these criteria, licensees of currently operating plants and applicants applying for operating licenses should be required to evaluate the size of the control room staff needed.

Conclusion 5 - Reporting of Incidents to Licensees

Documentation was not available to determine whether all licensees of operating nuclear plants receive timely reports of incidents that occur at other similar facilities indicating the probable cause of equipment malfunctions that occur. In addition, it could not be determined whether licensees have evaluated the potential for similar incidents in their own plants and, where appropriate, have taken or plan to take corrective action to minimize the probability of their occurrence.

Recommendation

It is recommended that a system be developed and implemented by AEC to promptly (within 10 days) and fully inform licensees of incidents or unusual occurrences. Field inspections by Regulatory Operations personnel should routinely include review of licensee plans for evaluating and taking corrective action on such incidents.

Conclusion 6 - Reporting of Incidents to AEC

Wide variations exist in the scope and depth of information reported to AEC by licensees regarding incidents. Sufficient information is not uniformly reported to evaluate incidents fully and independently.

Recommendation

It is recommended that the regulatory staff develop an incident reporting guide to supplement or modify Safety Guide 16.C.2.a. This should include information such as (1) reactor operating conditions and the operational status of systems prior to the incident, (2) the chronology of events, including sources of data (computer readout or indicator reading) and including notations of equipment or human error, (3) contributing factors, including an analysis of the cause of human errors, (4) maximum and minimum conditions during the transient, (5) post-incident actions, including audits or reviews conducted independently of the line operating organization, (6) quantity and composition of the radioactive materials released, (7) damage incurred to systems, components and structures, (8) plans and schedules for corrective actions (including comparison with technical specifications requirements and justification of the adequacy of actions), (9) plans and schedules for continuing investigations, and (10) date power generation was or is expected to be resumed.

Conclusion 7 - Staff Actions on Incidents and Occurrences

The scope of the regulatory staff review of the incidents and the basis for the decisions made could not be fully determined from the documentation available. Documentation of follow-up actions by the staff with respect to long-term licensee programs and commitments also was not generally available.

Recommendation

Regulatory policies and procedures should be revised to more clearly identify the responsibility for review, decision-making, investigation, and documentation with respect to incidents and unusual occurrences.

APPENDIX A

Members of the study group that participated in the evaluation are listed below.

Name	Organization
R. L. Cudlin	Auxiliary and Power Conversion Systems Branch, Directorate of Licensing
R. J. McDermott	Reactor Testing and Operations Branch, Directorate of Regulatory Operations
R. E. Muranaka	Boiling Water Reactors Branch #2, Directorate of Licensing (*)
R. D. Silver	Operating Reactors Branch #2, Directorate of Licensing
V. D. Thomas	Technical Assistance Branch, Directorate of Regulatory Operations
R. H. Vollmer	Quality Assurance Branch, Directorate of Licensing
T. R. Wilson (Chairman)	Office of Operations Evaluation Directorate of Regulatory Operations

(*)Temporary Duty Assignment from the Division of Nuclear Education and Training.

APPENDIX B

INCIDENT CHRONOLOGIES

INTRODUCTION

In this appendix, the detailed chronological sequence of events and other pertinent summary information and data concerning each of the incidents evaluated by the study group are presented. The order of presentation of the incidents is the same as that in the preceding report. Specific information and data presented for each incident consist of the following:

Incident Date
Plant License Date
Operating Conditions When Incident Occurred
Initiating Mechanism
Chronological Sequence of Events
Equipment Response
Operator Action
Conditions During the Incident (Max. and Min.)
Incident Consequences
Damage Assessment
Corrective Actions
Date Power Generation Resumed

1.0 DRESDEN UNIT NO. 2

(Docket No. 50-237)

Incident Date: June 5, 1970

License Date: December 22, 1969

Operating Conditions When Incident Occurred:

Power Level: 623 MWe - 75%
Reactor Pressure: 965 psig
Mode of Operation: Steady State

Initiating Mechanism:

Spurious signal in one of the pressure regulators in the Electro-hydraulic Control (EHC) unit of the turbine-generator caused the turbine control valve to open from 75% to 80%; simultaneously all turbine bypass valves opened fully and remained open for 22 seconds.

Sequence:

Total Elapsed Time	Event
1. Time zero (21:28:40)	Control valves opened from 75% to 80% power and the bypass valves fully opened resulting in high steam flow of about 115 percent rated flow.
2. 1 second	High steam flow sensors tripped, but primary system isolation did not occur in both channels simultaneously.
3. 1 second	The turbine tripped.
4. 1 second	The reactor scrammed automatically from turbine trip.
5. 2 seconds	Generator load rejection occurred.

6.	3 seconds	All four low water level sensors tripped from low reactor vessel water level (20" on Yarway).
7.	5 seconds	After going into the "pump runout," the two operating feed pumps "B" and "C" tripped automatically.
8.	7 seconds	Reactor feed pump "C" restarted automatically.
9.	19 seconds	Water level increased from below the low level setpoint to more than 55 inches on the GE/MAC indication.
10.	22 seconds	The turbine bypass valves were fully closed automatically.
11.	33 seconds	Main steam line low pressure trips (850 psig) occurred in both isolation channels causing an isolation signal.
12.	33 seconds	The main Steam Isolation Valves (MSIV's) reached 10% closed.
13.	35-40 seconds	Water level decreased from more than 55 inches to less than 15 inches on the computer trend chart.
14.	50-75 seconds	Water level rose from less than 15 inches to more than 55 inches on the computer trend chart, but stuck at about 17 inches on the GE/MAC recorder chart.
15.	1 minute	Reactor pressure decayed to 775 psig and began rising.
16.	1 minute 30 sec - 2 minutes	Operator discovered stuck water level recorder pen on GE/MAC indicator and manually reduced feedwater flow to about 2.7×10^6 lb/hr.

17.	2-3 minutes	Operators attempted to bring isolation condenser (IC) into operation and to open the MSIV's. Neither attempt was successful.
18.	3 minutes 45-50 sec.	Reactor pressure peaked at 1054 psig (later determined to be 1097 psig) and began decreasing rapidly from manual operation of one electromatic relief valve.
19.	5 minutes 38-43 sec.	Reactor pressure fell to 960 psig and began rising.
20.	5-6 minutes	Safety valve "G" on main steam line "D" opened momentarily and the impingement of steam cocked the lifting levers of valves E and F.
21.	6 minutes 3-8 sec.	Pressure increased to 1050 psig and began falling rapidly from manual operation of one relief valve.
22.	6 minutes 3 sec.	Drywell containment high pressure trip actuated (2 psig).
23.	6 minutes 7 sec.	The diesel generator cooling pumps started automatically.
24.	6 minutes 12 sec.	Both recirculation pumps tripped automatically.
25.	6 minutes 13 sec.	Both emergency diesel generators started automatically.
26.	6 minutes 24-30 sec.	Both core spray systems started automatically, but did not inject any water into the vessel.
27.	About 6 min. 30 sec.	The high pressure coolant injection (HPCI) system received a start signal.
28.	6 minutes 35 sec.	All four low pressure coolant injection (LPCI) pumps started automatically but no water was injected into the vessel.

29.	8 minutes 10-15 sec.	Reactor pressure fell to about 840 psig and began rising.
30.	9 minutes 15 sec.	1/3 of the bypass valves opened. (They reclosed again at 11 mins. 16 secs.)
31.	9 minutes 31-36 sec.	Pressure peaked at about 1097 psig and began decreasing when two relief valves were opened manually.
32.	9 minutes 45 sec.	The earlier IC isolation was reset and the IC was initiated manually.
33.	13 minutes 8 sec.	The first IRM erratic behavior was printed out by the process computer.
34.	13 minutes 1 sec.	The first LPRM system erratic behavior was printed out by the process computer.
35.	14 minutes	The MSIV's were opened manually.
36.	20-21 min.	Reactor feed pump "C" was tripped manually.
37.	About 30 mins.	Containment temperature indicators read by operator, highest temperature indication was 205°F.
38.	30-40 mins.	Core spray and LPCI pumps were shut down.
39.	30-40 mins.	Emergency diesel generators were stopped and SBGTS was opened to the drywell.
40.	45 minutes	Shutdown cooling was shifted to the main condenser via a bypass valve.
41.	1 hr. 2 min.	Drywell floor drain sump pump started automatically.
42.	1 hr. 15 min.	Five of seven drywell coolers were brought manually into service.

43.	1 hr. 30 min.	LPRM power suppliers were shut off.
44.	2 hours	The drywell pressure indication was back on scale and read about 2.2 psig.
45.	3 hrs. 30 min.	A composite particulate filter sample from the drywell was measured to be 5 rad.
46.	5 hrs. 20 min.	A stack gas charcoal filter, removed at 4 hours, was analyzed for iodines.
47.	6 hours	A particulate filter sample (sampling time about 1 hour) was taken.
48.	9 hrs. 30 min. (0700 hrs. 6/6/70)	The particulate filter sample above was re-analyzed for iodines.
49.	0955, 6/6/70	Operations and radiation protection personnel entered the drywell.
50.	1030, 6/6/70	Reactor water level was lowered below the main steam nozzles.
51.	1300-1600, 6/6/70	Radiation protection personnel analyzed onsite air samples and onsite grass samples. No significant amounts of iodine were found.
52.	2020, 6/6/70	A Staplex air sample from the drywell was taken.
53.	2300, 6/6/70	The drywell purge rate was increased.
54.	0600, 6/7/70	A Staplex air sample from the drywell showed the I-131 level to be down to about 25 maximum permissible concentration (MPC).
55.	100, 6/7/70	Operations and radiation personnel entered the drywell for an inspection.

Equipment Response:	Water level recorder pen stuck. Isolation condenser operation difficulty. Misdirection of safety valve discharge and failure of two safety valves to close. Failure of LPRM cables. Pressure regulator failure. Diesel generators and their cooling pumps started automatically (normal). HPCI received initiation signal but was inoperative prior to the incident. Relief valve manual operation was normal.
Operator Action:	Operator switched feedwater control system to manual but failed to control water level. Operated relief valves to control system pressure (normal). Vented containment in violation of operating procedures. Failed to operate containment spray pumps to reduce drywell pressure (violation of operating procedures). MSIV's manually opened (normal). Shutdown core spray and LPCI pumps (normal). Stopped diesel generators (normal). Opened SBGTS to the drywell above design pressure of SBGTS (violation of operating procedures). Manually started drywell coolers (normal). Isolation condenser brought into service (normal).
Maximum and Minimum Conditions During Transient:	Drywell Pmax: 20 psig (calculated). Drywell Tmax: 320°F (calculated). Reactor Pmax 1097 psig. Reactor Pmin: 775 psig.
Incident Consequences:	Core never exposed. No measurable release of radioactivity to the environment. Drywell pressurized.

Damage Assessment:	Paint peeled off drywell wall. IRM, SRM, LPRM cables damaged. Safety valves lifting levers deformed. Thermal insulation damage. Grounds in electrical equipment. No fuel damage. Drywell cooling blower motor shorted.
Corrective Actions:	Procedures revised for venting containment per management approval. Installation of new IRM, SRM and LPRM cables. High range drywell pressure instrumentation made operable. Safety valves replaced with steam tested valves. Re-orientation of safety valve discharge. Field test to show ECCS sequencing was proper. Rupture discs downstream of safety valves replaced. New ceramic potentiometer installed. Electrical penetration repotted with sealing compound. Drywell cooling blower motor replaced. Modified feedwater control valve. Reset isolation condenser trip.
Date Power Generation Resumed:	August 11, 1970 Downtime: 52 days

2.0 MONTICELLO UNIT NO. 1

(Docket No. 50-263)

Incident Date:	September 28, 1971
License Date:	September 8, 1970
Operating Conditions When Incident Occurred:	Power Level: 327 MWe - 60% Reactor Pressure: Normal Mode of Operation: Steady state
Initiating Mechanism:	Recirculation pump control malfunction.

Sequence:

Total Elapsed Time	Event
1. Time zero (0125)	Reactor scram and isolation.
2.	"D" safety/relief valve manually opened and closed to maintain reactor pressure between 950 psig and 1050 psig.
3.	"D" safety relief valve manually actuated for second time.
4.	"D" safety/relief valve failed to reseat following return of control switch to "close" position.
5.	Reactor pressure continues to decrease.
6. 37 seconds	"D" safety/relief valve closes after receiving another open and close signal from a manual switch in the the control room.

Equipment Response:	Relief valve fails to reseat.
Operator Action:	Manual operation of safety/relief valve (normal).

Maximum and Minimum Conditions During Transient:	Reactor Pmax: 1040 psig. Reactor Pmin: 670 psig Core never exposed.
Incident Consequences:	No significant release of radioactivity reported. No pressurization of drywell.
Damage Assessment:	Subsequent investigation revealed stems on the safety/relief valve main discs to be galled.
Corrective Actions:	Only immediate action was for the "D" safety/relief valve handswitch to be tagged, with use of other safety/relief valves for manual pressure control in the event of reactor isolation. Manual control circuit for "D" safety/relief valve was subsequently tested and found to be normal. New set pressure adjusting springs, and new main discs, with stellite coated stems, installed; inside edges of stem bushings chamferred, and insulation removed for the four safety/relief valves. Testing of repaired valves. Testing and adjustment of Recirculation Pumps controls.
Date Power Generation Resumed	September 28, 1971 Downtime: 22 hours

3.0 MILLSTONE 1 UNIT NO. 1

(Docket No. 50-245)

Incident Date:	October 10, 1971
License Date:	October 7, 1970
Operating Conditions When Incident Occurred:	Power Level: 652 MWe - 100% Reactor Pressure: 1035 psig. Mode of Operation: Steady State
Initiating Mechanism:	Spurious signal in electric pressure regulator causing several turbine condenser by-pass valves to open.

Sequence:

Total Elapsed Time	Event
1. Time zero (2104)	Instability of the electric pressure regulator causing one turbine by-pass valve to open fully and a second valve to open 60%. The operator, in an attempt to regain control, switched to the mechanical pressure regulator (MPR) which did not improve conditions (normal).
2. 3 minutes	The operator reduced reactor load via the recirculation pump speed control. This step is in accordance with operating procedures.
3. 4 minutes	Reactor scram, initiated from high flux in APRM/Flow Bias owing to the rapid reduction in core flow (Step 2) which was under way. Reactor pressure, which causes the high flux trip, terminated at 1040 psig.
4. 4 minutes +	Turbine tripped - manually. However, the fully open by-pass valve remained fully open.

5. 6 minutes	Main Steam Isolation Valves (MSIV's) closed by operator. Reactor pressure blowdown continued, and the operator found that the "B" Automatic Pressure Relief (APR) was open.
6. 22 minutes	APR "B" closed 18-minutes after scram. Reactor temperature - 390°F; pressure, 263 psig.
7. 39 minutes	Reactor temperature - 430°F; pressure, 360 psig. The Plant was then put into a normal controlled cooldown using the isolation condenser.
Equipment Response:	Turbine control system instability. Electromatic relief valve lifted prematurely (1040 psig) and did not reseat within the 8 to 10% of its set point (1095 ± 12 psig). Isolation condenser functioned normally. ECCS was not initiated as set point conditions were not reached.
Operator Action:	Turbine control system was switched from electric to mechanical pressure control (normal). Reduced reactor load by reducing recirculation pump speed in accordance with operating procedures. Tripped the turbine (normal). Closed the MSIV's (normal). Initiated the operation of the isolation condenser for controlled cooldown (normal).
Maximum and Minimum:	
Conditions During Transient:	Reactor Pmax: 1040 psig Reactor Pmin: 264 psig Reactor Tmax: 525°F Reactor Tmin: 390°F

Incident Consequences:	Core never exposed. No measurable release of radioactivity to the environment. Drywell did not pressurize. The transient which occurred was less severe than that given in the design specifications, i.e., this transient is negligible with respect to vessel life. A total of 75,000 gallons of water was discharged to the torus during the time from the incident through reactor cooldown.
Damage Assessment:	"B" relief valve was disassembled, and it was found that the main disc was "wire drawn," and the disc of the pilot valve was badly "scored." A broken line to the pilot valve was found. Review of the turbine control system revealed loose linkages, and disconnected dash pot.
Corrective Actions:	The relief valve set pressure adjustment springs were found to be operating in ambient conditions greater than intended. This resulted in relaxation of the springs which in turn caused one of the relief valves to lift prematurely. The springs were replaced with a type which is better suited for this installation and operation. Insulation was removed from the upper portion of valves to prevent high operating temperatures. Valve seats and discs were redressed and looped as needed. All relief valves were checked for proper lift pressures prior to operation. Appropriate modifications were made to the turbine control system, resulting in a more reliable system.

Weekly tests of the turbine control system during power operation will be performed.
A study made of the effect of the blowdown uncovered no significant problems.
Inservice Inspection Program performed on a continual basis.

Date Power Generation Resumed: October 23-24, 1971
Downtime: 13 days

4.0 NINE MILE POINT UNIT NO. 1

(Docket No. 50-220)

Incident Date	December 31, 1971
License Date:	August 22, 1969
Operating Conditions When Incident Occurred:	Power Level: 473 MWe - 95% Reactor Pressure: 1015 psig. Mode of Operation: Steady State
Initiating Mechanism:	During routine monthly surveillance test on the reactor level instrumentation, the mechanic's wrench slipped and hit a companion instrument, apparently causing coincident high level signals that tripped the turbine and resulted in reactor scram.

Sequence:

Total Elapsed Time	Event
1. Time zero (1008:02)	A turbine trip occurred from an erroneous high reactor water level signal caused by bumping the sensors.
2. 0 seconds	Reactor scram from turbine trip.
3. 18 seconds, approximately	Shaft driven feedwater pump placed in manual control by operator.
4. 25 seconds, approximately	No. 12 motor driven feedwater pump placed in manual control by operator.
5. 28 seconds	Main steam isolation valves closed automatically.
6. 31 seconds	No. 11 motor driven feedwater pump placed in manual control by operator.

7.	1 minute 28 seconds	Reactor level plus three feet above normal.
8.	1 minute 58 seconds	Reactor pressure 1117 psig.
9.	2 minutes 9 seconds	Relief valve 121 open. Water flowed into the main steam lines.
10.	2 minutes 13 seconds	Relief valve 121 closed.
11.	2 minutes 54 seconds	Relief valves 111, 112, 122 open.
12.	2 minutes 57 seconds	Relief valves 112 closed.
13.	2 minutes 58 seconds	Relief valves 111, 122 closed.
14.	3 minutes 58 seconds approximately	Feedwater flow to 0
15.	11 minutes 58 seconds	Reactor water level under control.
16.	Same day	Station operating review committee reviewed occurrence and approved return to power.
17.	11:55 p.m. - December 31	Plant brought back on line.

Equipment Response: Relief valves functioned normally. Emergency condenser functioned normally.

Operator Action: Operator switched feedwater control to manual, but failed to control water level (water flowed into main steam lines). Emergency condenser placed in service (normal).

Maximum and Minimum Conditions During Transient: Reactor Pmax: 1117 psig.

Incident Consequences: Core never uncovered. Offgas spiked from 25,000 microcuries per second to 60,000 microcuries per second after turbine trip. No pressurization of drywell.

Damage Assessment:	No damage reported.
Corrective Actions	Expected feedwater system response was reviewed with operators.
Date Power Generation Resumed:	December 31, 1971. Downtime: Less than one day.

5.0 DRESDEN UNIT NO. 3

(Docket No. 50-249)

Incident Date: December 8, 1971

License Date : January 12, 1971

Operating Conditions When Incident Occurred:

Power Level: 972 MWe - 100%
Reactor Pressure: 990 psig
Mode of Operation: Steady State

Initiating Mechanism: Condensate booster pump tripped from unknown causes

Sequence:

Total Elapsed Time	Event
1. Time Zero (1413:08)	One of three operating condensate booster pumps tripped from unknown causes.
2. 1 second	Operating feedwater pumps (2) tripped due to low suction pressure.
3. 2 seconds	Standby feedwater pump started automatically. Full flow in approximately 6 seconds.
4. 2 - 14 seconds	Water level decreased.
5. 14 seconds	Reactor scrammed on low water level - 143" above the top of the active fuel, 0" on the GE-MAC scale.
6. 25 seconds	Reactor water level reached low point of - 20" (GE-MAC), or 123" above active fuel, and began rising.
7. 25 - 65 seconds	Operator reduced master feedwater controller set point; transferred control to manual; closed low flow feedwater valve.

		At a vessel water level of 12" (GE-MAC), the operator opened low flow feedwater valve due to level hesitation. With level increasing, operator closed low flow feedwater valve and attempted to close the feedwater isolation valve. This valve stalled due to high differential pressure but reduced flow from 5.7×10^6 to 2.3×10^6 lb/hr.
8.	1 min. 6 seconds	Group I isolation due to main stream line low pressure. Reactor pressure reached low point of 795 psig.
9.	2 minutes 7 seconds	Turbine tripped on high reactor water level. Operator put isolation condenser in service-ineffective due to high water level.
10.	2 minutes 45 seconds	Reactor water level reached level of main steam lines and began filling them.
11.	5 minutes	Safety valve 3F lifted at a reactor pressure of 1020 psig. for approximately 1.5 min.
12.	5 minutes 11 seconds	Reactor drywell pressure at 2 psig. ECCS activated, Diesel generators started.
13.	6 minutes 30 seconds	Drywell conditions peaked at 20 psig and 295°F.
14.	7 minutes	Torus cooling placed in service via recirculation through containment cooling heat exchangers.
15.	8 minutes	LPRM cable failures.
16.	13 minutes 5 seconds	Feedwater pump manually tripped.
17.	3 hours 47 minutes	Jumper installed to permit water sampling.

18. 6 hours 47 minutes	Reactor water blowdown established via cleanup system.
19. 12 hours	ECCS returned to standby condition. Drywell pressure at 1.75 psig.
20. 13 hours	Containment atmosphere sampled.
21. 43.8 hours	Drywell entered to obtain atmosphere samples.
Equipment Response:	Feedwater control valve failed to function due to an insufficient air supply. A safety valve lifted when system pressure was 200 psi below valve set pressure. Isolation condenser function ineffective due to high water level. LPRM cable failures. ECCS and diesel generators functioned as required.
Operator Action:	The operator failed to reset the feedwater control valve on lockout. Failure to trip feedwater pump when water level exceeded +60 inches. Isolation condenser put into service (normal) Torus sprays were not initiated in accordance with procedures.
Maximum and Minimum Conditions During Transient:	Drywell Pmax: 20 psig Drywell Tmax: 295°F Reactor Pmax: 1020 psig Reactor Pmin. 795 psig Reactor Water Level (min): 123 inches above fuel Reactor Water Level (max): main steam lines
Incident Consequences:	No measurable release of radioactivity to the environment. Core never exposed.

Damage Assessment:	LPRM cables damaged. Containment cooling fan motor (1) shorted. Electromatic relief valve solenoid (1) shorted disabling the valves. 125 volt DC grounds in drywell-annunciator panel lost as a result. Peeled paint off drywell wall.
Corrective Actions:	LPRM cables replaced. Relief valve solenoid replaced and tested. Safety valve replaced with a spare valve and horn discharge redirected. Feedwater control valve air supply system equipped with larger air supply line and an accumulator. Operating procedures modified. A program to investigate safety valve performance was initiated. A study of the feedwater control system was begun. Reactor operator retraining was commenced.
Date Power Generator Resumed	December 28, 1971 Downtime: 20 days

6.0 NINE MILE POINT UNIT NO. 1

(Docket No. 50-220)

Incident Date	February 28, 1972
License Date:	August 22, 1969
Operating Conditions When Incident Occurred:	Power Level: 476 MWe - 96% Reactor Pressure: 1011 psig Mode of Operation: Steady State
Initiating Mechanism:	Reactor protection system continuous power supply failed causing loss of electric power to half of the reactor protection system and part of the feedwater control system.

Sequence:

Total Elapsed Time:	Event
1. Time Zero (1331:31)	Low output voltage alarm on the M/G set.
2. 12 seconds	All sensors in channel 11 reactor protection system tripped in a fail safe mode on loss of voltage.
3. (approximately 1331)	Loss of feedwater control and feedwater valve lockup due to loss of the M/G set.
4. (approximately 1331)	Cleanup system isolated.
5. 14 seconds	Reactor scram due to low level trips.
6. 29 seconds	Main steam isolation valves closed due to reactor pressure of 850 psi in run mode.
7. 3 minutes	M/G set automatically transferred back to AC drive.

8.	8 minutes 20 seconds	Unsuccessful attempts to restore cleanup system by operator.
9.	9 minutes 44 seconds	M/G set automatically transferred to DC drive again, loss of continuous power supply voltage.
10.	11 minutes 52 seconds	Reactor level plus three feet above minimum normal water level.
11.	13 minutes 1 second	#111 electromatic relief valve opened at setpoint. Water flooded the main steam lines about this time.
12.	13 minutes 1 second	#112 electromatic relief valve opened at setpoint.
13.	13 minutes 3 seconds	M/G set auto transferred back to AC drive.
14.	13 minutes 5 seconds	#112 electromatic relief valve closed.
15.	14 minutes 58 seconds	#121 relief valve opened at setpoint.
16.	15 minutes 26 seconds	#111 relief valve closed.
17.	15 minutes 26 seconds	#112 relief valve opened at setpoint.
18.	15 minutes 31 seconds	#121 relief valve closed.
19.	15 minutes 31 seconds	#112 relief valve closed.
20.	15-1/2 - 42-1/2 min.	Repeated relief valve operation.
21.	18-1/2 minutes	Cleanup system restored to service by operator.
22.	42-1/2 minutes	Reactor level and pressure under manual control. Reactor level below emergency condenser nozzles.

23. 42-1/2 -52-1/2 min.	Manual operation of relief valve to reduce pressure.
24. 52-1/2 minutes	Emergency condenser placed in service by operator to control reactor pressure.
Equipment Response:	Transfer control circuit of the reactor protection system M/G set malfunctioned due to a blown fuse (6 amp fuse had been installed instead of the required 10 amp fuse). Relief valves functioned normally. Emergency condenser functioned normally. Feedwater valves locked in place preventing water level control. Cleanup system isolated preventing water removal.
Operator Action:	Cleanup system restored to service (normal). Emergency condenser placed in service (normal). Manual operation of relief valves to reduce pressure (normal).
Maximum and Minimum Conditions During Transient:	Reactor Pmax: not available.
Incident Consequences:	Core never exposed. No pressurization of drywell. No significant release of radioactivity.
Damage Assessment:	No damage reported.
Corrective Actions:	Time delay of the underfrequency relay was changed from 70 milliseconds to 110 milliseconds. Fuses in all M/G sets were checked. Will have surveillance of M/G set control panels to indicate a blown fuse. Feedwater lock-up circuits rewired so that not all circuits are powered from the same M/G set.

Date Power Generation Resumed: March 2, 1972
Downtime: 2 days, 14 hours.

7.0 MONTICELLO UNIT NO. 1

(Docket No. 50-263)

Incident Date: February 26, 1972

License Date: September 8, 1970

Operating Conditions When Incident Occurred:

Power level: 545 MWe - 100%
Reactor Pressure: 1007 psig.
Mode of Operation: Steady State

Initiating Mechanism: During routine testing of the turbine stop valves, rapid movement of one stop valve caused a reduction in pressure of the turbine emergency trip oil supply. This resulted in the closure of all turbine stop valves.

Sequence:

	Total Elapsed Time	Event
1.	Time zero (0530:02)	Reactor scram from closure of turbine stop valves.
2.	1 second	High reactor pressure scram signal received at 1062 psig. Relief valves and a safety valve opened at approximately this time. Drywell pressure reached 0.9 psig.
3.	1 second	Low reactor water level scram signal received at +9".
4.	7 seconds	High reactor pressure scram signal reset at 1050 psig.
5.	20-39 seconds.	Low reactor water level scram signal reset.
6.	2 min., 26 sec.	Turbine lockout relay trip -- reactor water level +48". Main steam bypass valves opened immediately thereafter.

7. 8 min., 58 seconds	Drywell pressure returned to normal.
8. February 27, 1972 - PM	Reactor operations resumed.
9. May 13, 1972	Drywell entered and minor damage observed to thermal insulation around safety valve discharge. Reactor operations resumed.
10. June 5, 1972	Reactor shut down and safety valve replaced with a spare valve.
Equipment Response:	A safety valve actuated below its set pressure. Turbine stop valve traveled at an abnormally fast rate.
Operator Action:	Failed to maintain reactor water level. Reactor water level increased to turbine trip level (+48") and above. Reactor recirculation pump speed manually reduced too far resulting in "scoop tube" lock. Unlocking procedure not followed and resulted in pump speed increase.
Maximum and Minimum Conditions During Transient:	Drywell Pmax: 0.9 psig (recorded) Drywell Tmax: Reactor vessel Pmax: 1118 psig
Incident Consequences:	Core never uncovered. No measurable release of radioactivity to environment. Drywell pressurized.
Damage Assessment:	Minor damage to thermal insulation. Cover knocked off cable tray from steam. No fuel damage.
Corrective Actions:	No immediate corrective action. Safety valve was not replaced until June 5, 1972.
Date Power Generation Resumed:	February 27, 1972 Downtime: approximately 1-1/2 days.

8.0

DRESDEN UNIT NO. 3

(Docket No. 50-249)

Incident Date: May 4, 1972

License Date: January 12, 1971

Operating Conditions:

Power Level: 800 MWe - 100%
Reactor Pressure: 990 psig
Mode of Operation: Steady State

Initiating Mechanism: Spurious scram signal.

Sequence:

Total Elapsed Time	Event:
1. Time Zero (0901)	Reactor scrammed from a spurious signal. Computer indicated a half scram condition on the "D" channel reactor low water level sensor. Recorder charts, computer recall, and discussions with operating personnel indicate normal operating parameters prior to the reactor scram.
2. 20 seconds	Reactor water level reached its lowest level of -1" (142" above the top of the active fuel) as indicated from the computer recall program. Control room recorder remained on scale and indicated +1" (GE-MAC scale).
3. 35 - 37 seconds	Two reactor feedwater pumps were manually tripped at 28" (GE-MAC) by the operator as the reactor water level was increasing rapidly. Water level continued to increase at a slower rate to a maximum of 53" (GE-MAC) which is 23" above normal, but 4" below the high pressure coolant injection and isolation condenser steam inlet lines and 54" below the main steam lines.

4.	1 minute.	Reactor pressure decreased due to minimum pressure of 828 psig due to the transfer of steam to the turbine and continued to add feedwater. Recorder chart indicated minimum pressure of 815 psig.
5.	5 minutes	Main steam isolation valves closed when reactor water level increased to +48" with the reactor mode switch in other than "run" position. Reactor pressure began increasing.
6.	9 - 10 minutes	Valve 1301-3, isolation condenser effluent, failed to open when the operator attempted to place isolation condenser in service. Another operator was dispatched to manually open the valve.
7.	12 minutes	Electromatic relief valve 1A opened automatically at a pressure of approximately 1110 psig. Normal opening is 1135 psig. The valve is estimated to have remained open approximately 15 seconds. Operation of the valve decreased reactor pressure to 1040 psig. At about this time a safety valve momentarily opened.
8.	12 min. 35 sec.	Drywell pressure of 2 psig initiated primary containment isolation and start of emergency core cooling systems. All systems operated as required. No vessel injection occurred. The High Pressure Coolant Injection (HPIC) system did not inject because the HPCI turbine had tripped as designed when the reactor water level increased to +48". Maximum drywell pressure reached was 2.5 psig.
9.	14 minutes	"B" electromatic relief valve was opened manually to maintain pressure below 1100 psig.

10.	25 minutes	Isolation condenser was manually placed in service to control reactor pressure. After manually positioning valve 1301-3 off its seat, the valve operated satisfactorily from the control room.
11.	12 hours	Drywell purge and deinerting was initiated after reactor water temperature had decreased below 212°F.
12.	14 hours	Initial drywell entry to obtain air samples and inspect electromatic safety valves was made using self contained breathing apparatus. Radiation exposure to personnel was reported to be 8 mr/hr for the thirty minute entry with the maximum radiation rate of 50 mr/hr at contact with recirculation piping. Source of drywell pressure confirmed to be from one safety valve which had relieved.
13.	18 hours	Second, drywell entry confirmed initial inspection and the "A" safety valve, located about 2 ft. upstream of the "A" electromatic relief valve had actuated. Damage from the discharge of the safety valve was very minor and was limited to grazing the mirror insulation of a safety valve in the adjacent "B" main steam line and opening the insulation below the "E" electromatic valve. Neither of the above valves appeared to be damaged from the steam jet.
<u>Equipment Response:</u>		Isolation condenser failed to operate. Safety valve actuated about 110 psig below the intended setpoint (1220 psig). Automatic actuation of the relief valve was below design setpoint. MSIVs closed automatically at reactor water level of +48" with mode switch in other than run position (normal). ECCS operated as required.

Operator Action:	Feedwater pumps were tripped manually at 28" (normal). Relief valves operated manually to control system pressure (normal). Isolation condenser placed in service (normal). Drywell purge and deinerting according to operating procedures.
Maximum and Minimum Conditions During Transient:	Drywell Pmax: 2.5 psig Drywell Tmax: 185°F Reactor Pmax: 1110 psig Reactor Pmin: 888 psig Reactor Water Level max: +53" (GE-MAC) Reactor Water level min: +1" (GE-MAC)
Incident Consequences:	Core never exposed. No significant release of radioactivity. Drywell pressurized.
Damage Assessment:	Ruputure discs on 3 or 8 safety valves were ruptured from "burping" of the valves. Minor insulation damage.
Corrective Actions	Repair and testing of isolation condenser valve. Replacement of safety valve diaphrams. Replace safety valve that lifted with a spare valve. Testing of relief valves that were operated. Revised the operating procedure for feedwater pump trip.
Date Power Generation Resumed:	May 8, 1972 Downtime: 4 days

APPENDIX C

INTRODUCTION

The analysis performed by the Study Group of three incidents that occurred at Dresden Unit 2 and Dresden Unit 3 nuclear plants involving the release of coolant are presented in this appendix. The performance of individual plant systems to operate during the incidents has been critically evaluated.

In carrying out this evaluation, the functional performance requirements documents in the Final Safety Analysis Report and the Technical Specifications were reviewed and the response of the plant systems and their components during the incidents were evaluated against these requirements. Particular consideration was given to failures of systems and components to perform their intended functions and to factors contributing to improper equipment performance.

In addition to analysis of the performance of plant systems, control room operations, control panel instrumentation and controls, plant damage, corrective actions taken by the licensee, documentation, and the significance of radioactivity released with respect to public health and safety were considered and are discussed in this appendix.

No description of systems or components is presented herein. The reader is referred to the detailed descriptions provided in the FSAR and supporting documents.

APPENDIX C

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1.0 Dresden Unit 2 Incident, June 5, 1970

This incident was initiated when the reactor was operating at 75% of the power level at near equilibrium conditions. A spurious signal in the Electro-Hydraulic Control (EHC) unit of the turbine generator set caused the turbine control valves to open from 75% to 80% and caused all 9 of the turbine bypass valves to open fully. The initiating cause of the controller malfunction was determined to have been a spurious electrical signal that occurred in a potentiometer located in a pressure regulator of the EHC unit. This event resulted in a turbine trip and a reactor scram.

1.1 Performance of Plant Systems and Components

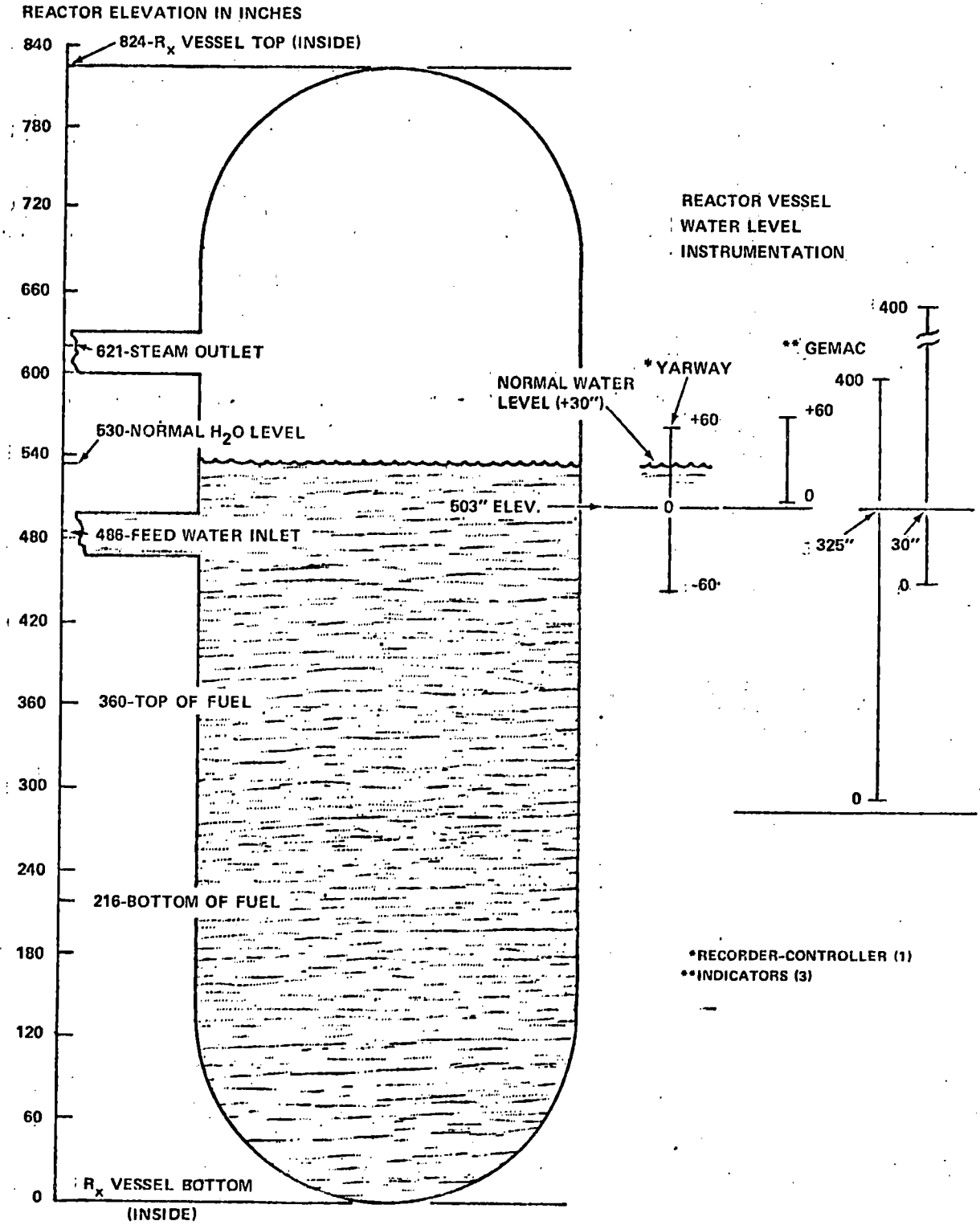
1.1.1 Feedwater System

1.1.1.1 System Requirements

The design objectives of the Feedwater System for Dresden Unit 2 at the time of the incident were as follows:

a. The feedwater system should maintain the reactor vessel water level within ±5" from the normal level (+30") during steady state operation in the automatic level control mode. Following abnormal occurrences at significant power that result in reactor scram and turbine trip, the Feedwater Control System was intended to maintain automatically the reactor vessel water level within the range of the 0 - 60" level recorder. Figure 1 indicates schematically the correlation between the reactor vessel water level and the instrument readings. The control functions performed by the water level instrumentation are as follows:

Figure I DRESDEN UNITS 2 AND 3 REACTOR VESSEL WATER LEVEL CORRELATION CHART



Control Functions Performed by Water Level Instruments

Control Function	Indicated Water Level on Yarway Instrument	Recorded Water Level on GEMAC Instrument
Turbine Trip	+55"	
MSIV Closure	+48"	
HPCI Turbine Trip	+48"	
FW Pump Trip		+48"
Auto Transfer of FWC from Flow to Level Control Mode		+30"
Low Water Level Scram	+1"	
Close Primary System Isolation Valves Except Main Steam Lines	+1"	
Initiate HPIC	-59"	
Low Water Level Recirculation Pump Trip	-59"	
Close MSIV's	-59"	
Start Diesels	-59"	

b. When the flow exceeds 12 x 10^6 lb/hr for two pump operation or 6 M lb/hr for one pump operation and the reactor level is below +30", the pump control transfers from the level control mode into the automatic flow control mode supplying 10.4 x 10^6 lb/hr with two pumps operation or 5.2 x 10^6 lb/hr for one pump operation. The system transfers back to the level control mode automatically when the level reaches +30". Manual reset is also possible.

c. When the air system pressure to the control valve drops to 75 psig or less, the control valve should remain in its existing position (lockout). The condition can be manually reset when air pressure is restored.

d. The feedwater pumps are to be able to be tripped manually when the reactor water level reaches +60".

1.1.1.2 System Performance

Following the initial water level drop as a result of the reactor scram, both operating feedwater pumps went into the flow control (runout) mode and tripped on low suction pressure. One of the pumps immediately restarted, as called for by the design, and provided water to the reactor vessel. During the

first minute following the scram, the water level varied between approximately +15" to +55" in response to inherent changes in the water level caused by void collapse and reheating.

Approximately 50 seconds after, the feedwater system was responding to a below-normal water level signal providing water to the reactor vessel. Although the water level had returned to the normal level, the operator continued to believe that a low level existed because of a stuck pen on the water level controller-recorder. He instituted manual control and increased feedwater flow thereby increasing the water level in the reactor vessel sufficient to cause flooding of the main steam lines within about 1-1/2 minutes.

1.1.1.3 Evaluation of System Performance

The initial fluctuations of reactor water level following scram are the result of changes in pressure, coolant inventory, decay heat level, and feedwater addition rate. The feedwater system was tracking level properly; however, the system pumping capacity was large enough to result in excessive water makeup flow into the reactor compared with shutdown makeup flow requirements. In addition, it was later determined that one of the feedwater control valves had a measured leak rate of 500 gpm which resulted in continued water addition to the reactor. Flooding of the main steam line and the line to the HPCI turbine may have contributed to premature safety valve operation, reduced the effectiveness of the isolation condenser and would have prevented operation of the HPCI system had it been operable and called on to function.

1.1.2 Overpressure Protection (Relief and Safety Valves)

1.1.2.1 Systems Requirements

The Technical Specifications for safety and relief valves require the following:

(a) The safety valves be set as follows:

2 valves at 1210 psig
2 valves at 1220 psig
2 valves at 1230 psig
2 valves at 1240 psig

with an allowable setpoint error of ±1%.

(b) The relief valves be tested during each refueling outage with set pressures as follows:

1 valve at 1125 psig
2 valves at 1130 psig
2 valves at 1135 psig

(c) All 8 of the safety valves and at least 4 of the 5 relief valves be operable during plant operation.

(d) A minimum of 4 of the safety valves be tested during each refueling outage.

The relief valves are sized by assuming a turbine trip with simultaneous scram, but with a failure of the turbine bypass system. Under these conditions the relief valve setting is chosen to be sufficiently low to eliminate the need for safety valve operation.

The actuation of relief valves is assumed to occur following events wherein the reactor is isolated from the main condenser or when electrical power is unavailable to pump water to the main condenser. During such events the core is cooled by the use of the isolation condenser system following depressurization through the relief valves.

1.1.2.2 System Performance

Following the turbine trip and reactor scram, system pressure decreased to 850 psig at which time the main steam isolation valves closed, as required and as designed. Reactor pressure continued to decrease to a minimum value of 775 psig and then began to increase.

Reactor pressure increased to 1050 psig (3 minutes 45 seconds after the scram) and the operator reduced system pressure to 960 psig (5 minutes 8 seconds) by operation of the "E" electromatic valve. At a time between 5 and 6 minutes after the scram, the "G" safety valve and possibly others lifted prematurely.

System pressure reached 1050 psig (6 minutes) and was again reduced by the operator opening the "E" electromatic relief valve, reducing pressure to 850 psig before closing the valve. Pressure again increased at 200 psi/min as a result of the combined effects of decay heat and compression of steam caused

by continued feedwater addition. Pressure rose to 1097 psig (9 minutes 31 seconds) and relief valves "D" and "E" were manually operated for 1 minute. Following this actuation of the relief valves, reactor pressure was controlled by operation of the isolation condenser, and by reopening the main steam line isolation valves and by use of the main condenser as a heat sink. The "E" and "F" safety valves were later determined to have opened and remained partially open due to impingement of the steam and water discharge from "G" safety valve onto their operating handles.

1.1.2.3 Evaluation of System Performance

During the incident, the pressure relief valves were not required to automatically actuate and their manual actuation was satisfactory.

The safety valves, whose set pressure is significantly (245 psig) above any recorded system pressure reached during the transient, should not have opened. However, the "G" safety valve apparently opened as a result of either mechanical shock or pressure transient when a relief valve was operated. The operator's recall of the event, however, did not support this conclusion because the operator recalled that only the "E" relief valve (which is located on the "B" steam line) had been used prior to the lifting of the safety valve (located on the "D" steam line). It is possible that due to the close proximity of the operating switches for the "D" and "E" relief valves, the "D" relief valve (located on the "D" steam line) may have been actuated.

The cause of premature actuation of the safety valve is as yet unknown; however, its operation breached the primary coolant system pressure boundary which is one of three barriers to the retention of radioactive materials. This aspect of the incident, the most significant in terms of damage and safety significance, was in part also caused by a design deficiency, i.e., allowing the safety valve discharge to strike other valves.

1.1.3 Containment System

1.1.3.1 System Requirements

Primary containment automatic isolation signals required by Technical Specifications are as follows:

Classification	Description of Signal	Max. Closure Time (Seconds)
Group 1	1. Reactor low-low water level	5
	2. Main Steamline high radiation	
	3. Main steamline high flow	
	4. Main steamline tunnel high temperature	
	5. Main steamline low pressure	
Group II	1. Reactor low water level	20
	2. High drywell pressure	
Group III	1. Reactor low water level	30
Group IV	1. HPCI steamline high flow	25
	2. High temperature near HPCI steamline	
	3. Low reactor pressure	
Group V	1. Isolation condenser high steam flow	30
	2. Isolation condenser high condensate flow	

1.1.3.2 System Performance

During the incident the containment isolated on main steamline low pressure (Group I) and high drywell pressure (Group II). It was calculated that the primary containment was subjected to a pressure of 20 psig and temperature of 320°F, although there were no operable wide-range pressure or temperature recorders to record the drywell pressure and temperature.

1.1.3.3 Evaluation of System Performance

The calculated temperature was in excess of the design value; however, the licensee has shown by analysis that a temperature of 320°F at 20 psig is less severe on the primary containment vessel liner than the design conditions of 281°F at 62 psig. Response of the containment to incident events was normal.

Containment leak rate testing, on June 13, 1970, of electrical penetrations indicated seven penetrations with significant leaks. It is possible that the degradation of the penetrations resulted from the environmental conditions (temperature, pressure and humidity) resulting from the incident. Containment system performance otherwise appears to have been normal and visual inspection revealed no indication of buckling of the containment liner. Local leak rate testing of containment isolation valves in May 1972, disclosed that four isolation valves were leaking well in excess of Technical Specification limits. The licensee considered that the elevated containment temperatures caused by the release of coolant may have contributed to the valve leakage experienced.

Leakage of the containment is of safety significance, since it is intended as a barrier against the release of radioactive materials. Although not a factor during the incident, the leakage experienced through the containment following coolant release incidents raises questions concerning the long term integrity of materials and forms the basis for the conclusion that the adequacy of containment components and materials exposed to coolant releases warrants further consideration.

1.1.4 Reactivity Control System

1.1.4.1 Systems Requirements

The system is designed such that the core can be made subcritical with the strongest control rod fully withdrawn. Technical Specifications require that the average insertion time for all operable rods from the fully withdrawn position to 90% full insertion be no greater than 5 seconds.

The following signals initiate automatic scram:

1. High neutron flux
2. High reactor pressure
3. High drywell pressure
4. Low reactor water level
5. Main steamline isolation

6. Main steamline high radiation
7. Generator load rejection
8. Turbine stop valve closure
9. Turbine condenser low vacuum
10. Loss of turbine control air pressure
11. High water level in scram discharge tank

1.1.4.2 System Performance

The reactor was scrammed automatically by the closure of the turbine stop valve. Nuclear instrumentation recordings and rod position indications verified that the reactor was shut down, and that the performance was within the limits of the Technical Specifications.

1.1.4.3 Evaluation of System Performance

Although cable damage caused a considerable number of the SRM, IRM and LPRM flux monitors to fail, there was never any question about the reactivity status of the reactor because it was more than 13 minutes after shutdown before any of the flux monitors began to fail and other instrumentation was available.

1.1.5 Electrical Systems

1.1.5.1 System Requirements

The Technical Specifications require the following:

(a) Two sources of off-site auxiliary power during operation.

(b) A minimum of one diesel generator per unit to be operable. The diesel is required to start automatically under conditions of high drywell pressure (2 psig), reactor low-low water level (83" above top of active fuel), and under voltage on emergency busses.

(c) Both 125 volt batteries be operable during plant operation.

1.1.5.2 System Performance

Following loss of the main generator output after the scram, the auxiliary loads that were being supplied by the generator through the auxiliary power transformer were transferred automatically to the reserve auxiliary power transformer. Both diesel generators for the unit started as designed upon receipt of the 2 psig high drywell pressure signal. The diesel

generators were not required and did not assume load because normal auxiliary power was available. The diesel generators were manually stopped 30-40 minutes after the scram.

1.1.5.3 Evaluation of System Performance

The Electrical System performed as designed. The manual stopping of the diesel generators in a manner that prohibited restarting automatically was in violation of the Technical Specifications. Although not of significance to safety during this incident, circumstances could have arisen necessitating restarting of the diesel generators automatically to allow safety equipment to perform its intended function. This reflects the shortcoming of procedures.

1.1.6 Emergency Core Cooling Systems

1.1.6.1 System Requirements

The Technical Specifications require:

(a) That the core spray and Low Pressure Coolant Injection (LPCI) subsystems be operable whenever irradiated fuel is in the reactor vessel. Exceptions to this requirement are permitted for specified lengths of time provided certain other testing is performed. Automatic starting of the LPCI and core spray subsystems is required for conditions of reactor vessel low-low water level* (83" above top of active fuel) or high drywell pressure (2 psig).

(b) That the High Pressure Coolant Injection system (HPCI) be operable whenever irradiated fuel is in the reactor vessel and reactor pressure is greater than 90 psig. Exceptions to this requirement are permitted for specified lengths of time provided certain other testing is performed. Automatic starting of the HPCI subsystem is required for conditions of reactor vessel low-low water level (83" above the top of active fuel) and high drywell pressure (2 psig).

(c) That 4 of the 5 relief valves that comprise the automatic pressure relief subsystem are required for conditions of reactor low-low level (83" above top of active fuel) and high drywell pressure (2 psig), 120 sec. time delay, and low pressure core cooling interlock.

*In conjunction with low reactor pressure of 300 to 350 psig.

1.1.6.2 System Performance

Approximately 6 minutes after the scram a 2 psig high drywell pressure trip signal was received. This automatically started both core spray pumps and the 4 LPCI pumps. The HPCI also received an initiating signal, but was valved out of service at the time and did not start. The HPCI turbine would not have functioned even if it had been operable because of the high reactor vessel water level signal that prevents the steam supply valve to the turbine from opening. In addition, the HPCI steam supply line was flooded at this time.

Injection of water into the reactor vessel from any emergency core cooling systems was not required and did not occur because reactor vessel pressure remained above the LPCI and core spray pump discharge pressure and the vessel water level was above the reactor core. The automatic pressure relief system was not initiated because the necessary and required trip signals were not both present. The ECCS systems which were activated were manually secured within 30-40 minutes when primary system pressure had decreased to less than 400 psig. This was done to prevent injection of torus water into the reactor vessel which would have occurred automatically at 300-350 psig.

1.1.6.3 Evaluation of System Performance

In consideration of the sequence of events during the incident and the initial condition of HPCI system (out of service), the emergency core cooling systems performed satisfactorily. However, even if operable, the HPCI system would have been unavailable for coolant injection had it been needed because of flooding of the HPCI steam supply line. Although other means are available for providing core cooling, since the HPCI system is intended to assist in providing core cooling in the event of a small line break, the HPCI system should be maintained in service to the maximum extent practicable and flooding of the inlet line to the HPCI turbine should be prevented.

1.1.7 Isolation Condenser

1.1.7.1 System Requirements

The Technical Specifications require:

(a) That the isolation condenser be operable whenever irradiated fuel is in the reactor vessel and pressure is

greater than 90 psig. An exception to this requirement is permitted for a specified length of time if the HPCI subsystem is operable. The system is required to be initiated automatically on a high reactor pressure of 1060 psig sustained for 15 seconds.

(b) That the system be automatically isolated under conditions of a break in the system.

(c) That the system be capable of removing decay heat production at 300 seconds following a scram.

1.1.7.2 System Performance

During the incident the main steamline isolation valve closed, as designed because of low reactor pressure. As pressure started to increase, the isolation condenser was manually actuated at a pressure well below its automatic actuation point. However, as the system was placed into operation, the initial surge of condensate return water caused the system to isolate on a high condensate flow signal.

Later during the incident the isolation condition was reset and the system placed manually in operation. However, since the system was flooded, its effectiveness was reduced.

1.1.7.3 Evaluation of System Response

The isolation condenser response during the incident was deficient because the instrumentation for the high condensate flow trip was incorrectly set. This resulted in a tripout when manual actuation of the system was first attempted. Even if the system had responded normally, its effectiveness would have been substantially impaired because of the flooding of the inlet lines. However, malfunction of the isolation condenser does not preclude cooldown and depressurization of the reactor vessel because of the availability of other systems to serve this function.

1.1.8 Reactor Coolant Cleanup System

1.1.8.1 System Requirements

The Technical Specifications require that the cleanup system automatically isolates on a reactor low water level signal (143" above top of active fuel).

1.1.8.2 System Performance

There was insufficient documentation available on the operation of this system.

1.1.8.3 Evaluation of System Performance

Insufficient documentation was available to assess performance.

1.2 Control Room Operations

1.2.1 Operating Personnel Qualifications*

Control room operating personnel on duty at the time of the incident were as follows:

Title	Type of License
Shift Engineer	SRO
Startup Engineer	SRO
Shift Foreman	SRO
Control Operator	RO
Center Desk Operator	RO
Control Operator (Unit 1)	RO
Shift Superintendent (General Electric Co.)**	SRO

All of the licensed operating personnel involved during the incident of June 5, 1970, had been assigned to the operation of Dresden Units 1 and 2 for several years. They had all received simulator training. The startup engineer had been assigned to Dresden Unit 2 for almost three years. The control room operator on duty at the time of the incident received his operating license for Dresden Units 1 and 2 in August of 1967, and August of 1969 respectively. The General Electric (GE) shift superintendent also held a senior operator license for Dresden 2 and had been at the station since September 1968.

*Plant operation responsibility is shown in Figure 6.1.2 of the licensee's technical specifications.

**General Electric Shift Superintendent was acting as an advisor to the licensee at the time of the incident.

1.2.2 Operating Deficiencies

1.2.2.1 Primary Containment

The containment atmosphere was vented to the environment prior to determining its radioactivity level and composition. In addition to the venting of the containment, water was pumped from the drywell sump to the radwaste system, and the drywell coolers were started. These operations are in noncompliance with Section 3.7.A.2 of the Technical Specifications which requires that containment integrity shall be maintained when reactor water temperature is greater than 212°F, and fuel is in the reactor.

The containment drywell sprays were not actuated when the pressure inside containment was greater than 2 psig, as required by the operating procedures. The decision not to use the drywell spray was prompted by a concern by operations supervision that further damage to equipment might be caused by operation of the sprays.

1.2.2.2 Emergency Core Cooling System

Thirty minutes after the scram, operations supervision directed that the core spray and LPCI pumps be placed in the "lockout" (shutdown) position before determining the cause of the incident. This action is considered in violation of the Technical Specifications which require operations involving nuclear safety to be carried out only in accordance with written procedures. Although the action taken was justified and had no safety significance under the circumstances associated with this incident, it reflects the inadequacy of written procedures for coping with abnormal occurrences.

1.2.2.3 Standby Gas Treatment System

The Standby Gas Treatment System (SGTS) was placed in operation when the drywell pressure was greater than that for which the SGTS was designed. This action was in nonconformance with the licensee's Final Safety Analysis Report, Section 5. Use of the SGTS did not result in a safety problem. Based on data obtained from area radiation monitors at onsite stations and environmental samples, no significant quantities of radioactive materials were released. Nevertheless, venting of the containment and use of the SGTS under conditions for which it was not designed reflect the lack of training and an attitude in conflict with good safety practices.

1.2.2.4 Feedwater Control System

At the time of the incident there were no written operating procedures available to operating personnel concerning how to handle an excessively high reactor vessel water level condition during a transient. Subsequently, these procedures (and others) were developed for both normal and emergency situations.

1.2.2.5 Reactor Vessel Water Level

Operations personnel confined their attention to one source of information with respect to reactor vessel water level, which gave an erroneous indication. Several other operable sources were at the same panel location.

Operations personnel acknowledged the "High Reactor Water Level" alarm but did not respond to correct the condition.

1.2.2.6 Procedures

Operating procedures were acknowledged to be deficient by facility management following the incident, in that they did not provide sufficient written instructions to the reactor operators for coping with an incident. An extensive and intensive operator training program supplemented by accurate and detailed operating procedures is a significant aspect of nuclear plant safety.

1.3 Control Panel Instrumentation and Controls

Prior to and during the incident, the wide range containment pressure monitoring system (0-75 psig) was inoperative. This instrumentation had been installed a short time prior to the incident and had not been placed in service. Also inoperative were the primary containment temperature monitoring systems which consist of two recording systems remote to the control room. During and subsequent to the incident, the isolation condenser system failed to operate because of the improper setting of the setpoint for initiating operation. The reactor water level recorder, which is part of the Feedwater Control System, experienced a temporary malfunction (pen "stuck" at 17 inches).

The lack of readily available information needed by the reactor operator to diagnose the status of plant systems and equipment

during transients raises questions concerning the adequacy of the control room design and is the basis for recommending a more extensive and intensive review of nuclear plant controls.

1.4 Plant Damage

The incident at Dresden 2 on June 5, 1970, during which steam was released to the torus by automatic actuation of one electromatic pressure relief valve and to the drywell by inadvertant actuation of three safety valves resulted in the following items of damage to the plant:

a. Insulation on the main steam line, two feedwater lines, and one recirculation riser was damaged by steam and water discharge from the safety valves and had to be replaced.

b. Six valve motors located in the drywell required drying to meet acceptable resistance levels. One (of seven) drywell cooling blower motor was replaced because of steam damage.

c. Control cables to the Traveling Incore Probe (TIP) were shorted and were replaced with a higher temperature rated cables.

d. Seven of the drywell electric penetrations were found to be leaking from the center inboard to the drywell and had to be repotted.

e. Four of eight intermediate range monitor (IRM) cables, two of four source range monitor (SRM) cables, and 103 of 164 local power range monitor (LPRM) cables were damaged and were replaced.

f. At two locations in the drywell, a one foot diameter patch of paint was removed by the safety valve discharge. This protective coating was replaced.

As a consequence of the lifting of the safety and relief valves, a total of approximately 250,000 pounds of coolant were discharged to the torus and the drywell, resulting in a drywell pressure increase of about 20 psig. The drywell pressure was vented to the Standby Gas Treatment System (SBGTS) after about 30 minutes, and the drywell pressure was thereby reduced to about 2 psig after two hours. Because of the primary coolant system activity at the time of the incident, fission product gases and particulates (Xe, Kr, I) as well as activation product gases and particulates were discharged to the drywell. The drywell venting was initiated at about 30 minutes and this activity was passed through the SBGTS absolute filter. Filter

activity increased from about 30 mr/hr to 50 mr/hr at the housing during venting. The Technical Specification limit for gross stack gas activity release is greater than 100,000 μCi/sec and the stack monitor is set to alarm at about 500,000 μCi/sec.

1.5 Corrective Action

Prior to return to operation, all damaged components (see 1.4 above) were repaired or replaced and the reactor returned to conditions required by technical specifications.

In addition, actions were taken to reduce the probability and consequences of a similar event. A ceramic potentiometer was installed in the electro-hydraulic control unit of the turbine generator set to avoid noise and spurious signals. Changes were made to the feedwater control system to improve level control. These changes included modifying feedwater control sensitivity, adjusting the main regulating valve to control on only one valve up to full load, making the small feedwater regulator valve control automatically at low flows on level control, and changing valve trim for better transfer to the large valve. Subsequent operating experience has shown that the corrective actions taken to improve water level control following this incident were not adequate and further study and corrective actions are in order.

Changes were made in the operating procedures to improve the control of water level during transient conditions, and to reduce the probability of a recurrence of the operator errors in this incident.

To minimize the damage resulting from steam discharge from a safety valve, the directions of all safety valve discharges were reoriented. The need for an extensive investigation of premature lifting of safety valves, however, was not identified by the licensee. Subsequent events at Dresden 2 and other similar nuclear plants has shown that further investigation and corrective action are in order.

1.6 Documentation

The criteria used as the basis for evaluating acceptability of documentation are presented in Section 3.7.

Following the incident, the licensee notified AEC within 24

hours by telephone, but did not also send a telegram, both of which are required by Technical Specifications. This apparently was an oversight.

Safety Guide No. 16 was not issued at the time of the June 5, 1970 incident. However, the licensee did submit a report on July 6, 1970, describing and analyzing the incident, evaluating the causes and discussing corrective actions. On July 27, 1970, the licensee submitted a supplemental report in reply to AEC questions. The latter report was received by the AEC about two weeks prior to return to operation. However, there is insufficient documentation of the basis for the conclusion that return to operation was acceptable, that the licensee's analysis of the incident was sufficient and that the short and long term corrective actions were acceptable.

During their review of the incident, the study group requested additional information to clarify the extent to which documentation of the June 5, 1970 incident had been made available to other similar operating plants.

We were informed that the reactor vendor sent reports of the June 5, 1970 Dresden 2 incident and December 8, 1971 Dresden 3 incident to Boiling Water Reactor owners who operate plants similar to Dresden 2 and 3 with the request that the reports be reviewed for implementation of actions or procedures that were applicable to their plants. Based on discussions with members of the staff, documentation was insufficient to determine that the staff was satisfied with followup investigations and corrective actions performed by licensees at other facilities.

1.7 Safety Significance

The significance of the June 5, 1970 incident at Dresden 2 was negligible with respect to the release of radioactivity on the health and safety of the public and plant operating personnel. Based on measurements obtained from onsite area radiation monitors and analysis of environmental samples for iodine, the activity level of radioactive materials released by venting of the drywell was significantly less than that allowed by the Technical Specifications during normal plant operation. No measurable increase above normal levels was noted and concentration levels were well below 10 CFR Part 20 limits.

The peak temperature in the drywell was measured to be 320°F and the peak pressure was estimated to be 20 psig. The design temperature and pressure are 62 psig and 280°F. The licensee

in a special report of the incident of June 5, 1970, demonstrated that the incident imposed less severe conditions on the drywell than those used in the design. The maximum primary system cooldown rate experienced during this incident was 125°F in 34 minutes with a vessel shell flange to shell differential temperature of 90°F or less. The Technical Specification limits for cooldown rate are 100°F per hour, averaged, or a step reduction of 240°F if the shell flange to shell differential temperature is less than 140°F.

Of particular importance to safety is the uncontrolled release of primary coolant from safety valves as a result of premature actuation and damaged lifting levers which could not be reseated through normal safety valve self-actuation or remotely by operator action. The steam release was, therefore, uncontrolled until drywell entry could be made to reseat the valves. The incident was, in this aspect, the same as a break in the primary system. During this transient, the Emergency Core Cooling systems responded but were not required for water addition.

Because the June 5, 1970 incident did not result in a release of radioactivity to the environs in excess of that allowed by the Technical Specifications during routine operation, and operable emergency core cooling systems functioned as designed, we conclude that this incident had minimal safety significance to the public. Although incidents of this type are undesirable, such incidents can be expected during the life of the plant. For this incident the plant demonstrated its capability for withstanding such events without undue hazard to the public in accordance with design intent. In view of the frequency of coolant release incidents, a question remains as to whether such incidents are acceptable when considering the number of plants to be placed in operation over many years. This question can be answered only by a realistic quantitative evaluation of accident probabilities and their consequences. Further, the difficulties encountered by the operators in understanding the condition of the plant throughout the transient and the capability to control easily all aspects of the pressure, temperature and water level changes raise the question of whether the plant instrumentation and controls are adequate to permit effective control by one or more operators in a reasonable manner.

2.0 Dresden Unit 3 Incident, December 8, 1971

This incident was initiated while the reactor was operating at 100% power at near equilibrium conditions, when the 3C condensate-booster pump tripped for unknown reasons. The loss of the 3C condensate-booster pump caused an automatic tripout of the two operating feedwater pumps because of low suction pressure. This event resulted in a reactor scram because of a low reactor vessel water level.

2.1 Performance of Plant Systems and Components

2.1.1 Feedwater System

2.1.1.1 System Requirements

Refer to System Requirements contained in Section 1.1.1 of Appendix B for Dresden 2, June 5, 1970 incident.

2.1.1.2 System Performance

Following the tripping of 1 of the 3 operating condensate-booster pumps, the two operating reactor feed pumps (3A and 3C) tripped because of a low suction pressure condition. The standby reactor feed pump "3B" started automatically as designed when the two operating feed pumps tripped. At this time, the "3B" feed pump went into a "runout" (110% feedwater flow - single pump 5.7×10^6 lbs/hr) or "flow control" condition as demanded by the feedwater control system. Reactor water level reached a low point of -20" (123" above the top of the active fuel) and then began to rise within 25 seconds.

The feedwater control valve failed in its existing position (lockout) because of insufficient air when the control system automatically reset from the runout condition. The control valve in its existing position allowed approximately 100% makeup water flow to continue to be added to the reactor vessel. The operators did not attempt to reset the "lockout" condition after it occurred and the reactor vessel water level reached the level of the main steam lines in approximately 3 minutes. The operator did attempt to close a block valve in the feedwater line to stop feedwater addition to the vessel, but the block valve did not fully close (flow reduced to 30%) because of a high differential pressure across the valve which tripped the valve motor. The operating feedwater pump was

tripped manually 13 minutes after the reactor scram and feedwater addition to the reactor vessel was terminated.

2.1.1.3 Evaluation of System Performance

The initial drop in reactor water level to the low level scram point was the expected result when the operating feedwater pumps tripped. The additional decrease in water level to -20" would be expected due to normal void collapse following the reactor scram. The standby feedwater pump started and injected water as designed; however, the feedwater control valve failed in its existing position which allowed the continued addition of feedwater flow at 100%. The valve failure apparently went unrecognized by the operator and resulted in a large overshoot in reactor vessel water level and caused flooding of the main steam lines. The cause for the valve failure was determined to be an air supply line to the feedwater control valve that was too small. The level overshoot could have been prevented by the early tripping of the operating feedwater pump, which was called for in operating procedures, but the feedwater pump was not tripped for 13 minutes. The feedwater control system was not able to control automatically the reactor water level and to maintain the level below the elevations of the isolation condenser, HPCI turbine steam supply lines and the main steam lines. The water level increase resulted in the flooding of these lines.

2.1.2 Overpressure Protection Systems (Relief and Safety Valves)

2.1.2.1 Systems Requirements

Refer to System Requirements in Section 1.1.2 of Appendix C for Dresden 2, June 5, 1970 incident.

2.1.2.2 System Performance

Following the reactor scram because of low reactor water level, the primary system pressure decreased as expected to 850 psig at which time the main steam isolation valves closed as designed (1 minute and 5 seconds after scram). Reactor pressure continued to decrease to a low of 795 psig (at 2 minutes) and then began to increase. An attempt was made to place the isolation condenser in service and control system pressure, but this was unsuccessful because of the high reactor water level.

Reactor pressure reached 1020 psig (5 minutes) at which time the 3F safety valve operated prematurely. It was estimated that the safety valve remained open for 1-1/2 minutes reducing system pressure to 910 psig. Two more pressure increases and decreases were experienced during the transient without additional safety valve operation; the first (at 13 minutes) whereon the system pressure was reduced from 1020 to 980 psig while the operating feedwater pump was turned off and the second (at 23 minutes) when a safety valve opened in the reactor water cleanup system and reduced pressure from 1050 to 950 psig. There was no indication that a relief valve had actuated during the transient.

2.1.2.3 Evaluation of System Performance

During the incident, the pressure relief valves were not required to actuate automatically and the valves were not actuated manually.

The "F" safety valve, whose set pressure was 220 psig above any recorded system pressure, opened prematurely from unknown cause(s). This aspect, the most significant in terms of plant damage and safety, has been postulated by the licensee to be the result of a design or equipment deficiency that has not yet been identified to date. An extensive program has been initiated by the licensee to investigate the cause of premature actuation.

2.1.3 Containment System

2.1.3.1 System Requirements

Refer to System Requirements contained in Section 1.1.3 of Appendix C for Dresden 2, June 5, 1970 incident.

2.1.3.2 System Performance

During the incident the containment isolated properly on low main steam line pressure (Group I) and high drywell pressure (Group II). It was observed that the primary containment was pressurized to 20 psi and experienced a maximum measured temperature of 295°F. Containment response does not appear to have differed from expectation. The calculated pressure increase resulting from discharge of the noncondensable gases from the drywell to the torus was about 20 psig. This agrees with the measured pressure indicated during the incident.

2.1.3.3 Evaluation of System Performance

The torus sprays were not turned on until 8 hours after the incident in violation of operating procedures that require torus sprays when the drywell pressure exceeds 2 psig. They were ineffective in reducing the pressure at this time. Significant pressure reduction would not be expected if the torus free volume were filled with noncondensable gases.

Drywell coolers were used to reduce containment pressure. Torus cooling through the containment heat exchangers was effective in maintaining torus water temperature below the Tech Spec limit of 95°F. The above cooling systems were activated manually.

Following the incident, the torus to drywell vacuum breakers were checked. The test arms on these check valves were actuated manually to determine freedom of movement. In three of the twelve vacuum breakers, the discs did not return to the closed position. Containment pressurization, similar to that which occurred during the incident could result in excessive containment pressure if the vacuum breakers had opened and failed to close.

2.1.4 Reactivity Control System

2.1.4.1 System Requirements

Refer to System Requirements contained in Section 1.1.4 of Appendix C for the Dresden 2, June 5, 1970 incident.

2.1.4.2 System Response

Reactor scrammed on low reactor vessel water level signal. The shutdown status of the reactor was verified immediately. Tests on 25 rods prior to the incident indicated an average time for 90% insertion time of 2.47 seconds.

2.1.4.3 Evaluation of System Performance

The system performed as designed and no problems were observed.

2.1.5 Electrical Systems

2.1.5.1 System Requirements

Refer to System Requirements contained in Section 1.1.5 of

Appendix C for Dresden 2, June 5, 1970 incident.

2.1.5.2 System Performance

Following loss of the generator output after the scram, the electrical loads that were being supplied by the generator through the auxiliary power transformer were transferred automatically to the reserve auxiliary power transformer. Both diesel generators started automatically as designed upon receipt of the 2 psig high drywell pressure signal. The diesel generators were not required and did not assume any of the station load because normal auxiliary power was available from off-site sources. The diesel generators continued to operate until the high drywell pressure initiating signal had cleared.

2.1.5.3 Evaluation of Performance

During the incident five grounds developed in the 125 volt d.c. circuitry within containment, which resulted in overload of the circuit for annunciator panel 903-3. The cause for the loss of power to control room annunciator panel 903-3 was attributed to the grounding of the controls for the "3A" electromatic relief valve that was damaged by the steam jet from the "3F" safety valve. Other circuit grounds were experienced on (1) the LPCI manually operated isolation valves 26A and 26B position indication circuits and (2) electromatic relief valves "3C" and "3D". No abnormal equipment operation occurred other than the loss of the annunciator panel. Based on the numbers of grounds experienced in the 125 volt d.c. system within containment, it does not appear that the system was designed or installed adequately to prevent failures or multiple grounds that could result in partial loss of d.c. power during incidents involving loss of coolant accidents. Safety related equipment supplied from the 125 volt d.c. system includes the pilot operators for the main steamline isolation valves and the autodepressurization system relief valves (a subsystem of the ECCS).

2.1.6 Emergency Core Cooling Systems

2.1.6.1 System Requirements

Refer to System Requirements in Section 1.1.6 of Appendix C for the Dresden 2, June 5, 1970 incident.

2.1.6.2 System Performance

Approximately 5 minutes after the reactor scram, a 2 psig high drywell pressure was received. This started both core spray pumps, the 4 LPCI pumps, and both diesel generators automatically. The HPCI also received an initiating signal, but at this time, the reactor vessel water level was in excess of the +48" level setpoint that causes automatic turbine trip and, therefore, the HPCI turbine did not start. Additionally, the line which supplies steam to the HPCI turbine was completely flooded with water at this time. Injection of water from any of the emergency core cooling pumps was not required and did not occur. There was no requirement for HPCI injection because the coolant inventory was well in excess of that required. The automatic overpressure protection system (relief valves) did not operate because the necessary and required signal of low-low reactor vessel water level coincident with a high drywell pressure signal was not present. The ECCS systems that were initiated during the incident were not secured until the high drywell pressure initiating signal had cleared 12 hours later.

2.1.6.3 Evaluation of System Performance

Operation of the emergency core cooling systems was initiated and the system performed in accordance with design.

The HPCI was not required to inject water during the transient. Flooding of the HPCI turbine steam supply line that was experienced created a condition in which the HPCI turbine was inoperable until the water could be drained from the HPCI steam supply line (estimated time 30 minutes). If restart of the HPCI turbine had been permitted, damage to the turbine may have resulted. The core spray could have been used as an alternate source of emergency core cooling water after reducing reactor vessel pressure by means of the relief valves.

Because the HPCI initiating signal of 2 psig drywell pressure was present, the HPCI turbine would have restarted automatically if the water level in the reactor vessel had decreased to approximately 45". To prevent auto restart of the HPCI turbine with the steam line flooded and possible damage to the turbine, an operator was dispatched to open the electrical breaker for the HPCI turbine steam control valve.

The operator failed to initiate the torus spray, as required by procedures, when the drywell pressure exceeded 2 psig because he correctly inferred from the containment pressure data available, that the sprays would have a negligible effect on the non-condensible gases carried over from the containment (drywell). The torus spray was not initiated until several hours after the event.

One of the 5 auto depressurization relief valves was disabled as a result of steam impingement from the safety valve that actuated. This resulted in only 4 relief valves being available to perform the depressurization function. A minimum of 4 operable relief valves is required by the technical specifications.

Numerous electrical grounds were experienced in the power supply required for operation of the auto pressure relief subsystem (125 volt d.c. system). These grounds did not, however, destroy the capability of the pressure relief system to function automatically. The loss of the capability of the pressure relief system to function automatically coupled with the loss of availability of the HPCI system as was experienced during this incident would remove ECCS capability for a small line break accident.

2.1.7 Isolation Condenser

2.1.7.1 System Requirements

Refer to System Requirements contained in Section 1.1.7 of Appendix C for the Dresden 2, June 5, 1970 incident.

2.1.7.2 System Performance

During the incident, the main steam line isolation valve closed because of low reactor pressure in accordance with the design intent. As pressure started to increase in the reactor, the isolation condenser was actuated manually at a pressure well below its automatic actuation point. The system was ineffective because its steam line was flooded with water.

2.1.7.3 Evaluation of System Performance

While the isolation condenser system valves initiated as designed, the system was not able to perform its function as a

heat sink with the main condenser isolated. The inability to control reactor water level resulted in flooding of the isolation condenser steam line, thereby reducing the effectiveness of the system to control the reactor pressure. However, as noted in section 1.1.7.3 of Appendix C, malfunction of the isolation condenser does not pose a safety problem.

2.1.8 Coolant Cleanup System

2.1.8.1 System Requirements

Refer to System Requirements contained in Section 1.1.8 of Appendix C for the Dresden 2, June 5, 1970 incident.

2.1.8.2 System Performance

During the incident the Reactor Coolant Cleanup System was isolated automatically on a low reactor water level trip at +8". The control room operator attempted to return the cleanup system to operation to provide a path for blowdown water from the reactor vessel because of high water level condition. Attempts to place the system back in operation were not successful. A blowdown through the Reactor Coolant Cleanup System safety valve (to the main condenser) resulted (23 minutes after the reactor scram) which reduced the reactor system pressure from 1050 to 950 psig, at which time the Cleanup System isolated automatically again for reasons unknown.

2.1.8.3 Evaluation of System Performance

The cleanup system isolated as designed.

Attempts to return the cleanup system to operation to provide a blowdown path and to reduce reactor water level were unsuccessful because of difficulty in controlling the system. The cause for the second isolation of the cleanup system could not be established, but may have been the result of one or more trip signals monitoring various cleanup system parameters.

2.2 Control Room Operations

2.2.1 Operating Personnel Qualifications

Control room operating personnel on duty at the time of the incident included:

Title	Type License
Shift Engineer	SRO
Startup Engineer	SRO
Shift Foreman	SRO
Control Operator	RO
Center Desk Operator	RO
Control Operator (Unit 1)	RO
Control Operator (Unit 2)	RO

Additional licensed operating personnel who were available during the incident included one shift foreman and several reactor operators.

All of the licensed operating personnel involved during the incident of December 8, 1971, had previous operating experience on Dresden Units 1 and 2. The Shift Engineer, who is a graduate engineer, was also on duty (in the same capacity) during the earlier blowdown incident that occurred at Dresden Unit 2 on June 5, 1970. The Control Operator had been licensed for approximately six months at the time of the Unit 3 incident, whereas the Center Desk Operator had received his operating license for Dresden Unit 1 and Units 2/3 in August 1967 and August 1969, respectively.

Subsequent to the December 8, 1971 incident, and as part of the scheduled retraining program, Operations Supervision reviewed with all licensed operators the latest operating and emergency procedures for: (1) excessively high reactor water level, and (2) drywell pressurization. Moreover, a simulator training refresher course was initiated (on a rotational basis) in January 1972 for all licensed reactor operating personnel. To date, approximately two-thirds (24 operators) of the licensed reactor operators have received this portion of the retraining program.

2.2.2 Operating Deficiencies

A. Feedwater Control System

Section 6.2.A of the facility Technical Specifications requires that approved written procedures for control of

feedwater and reactor water level transients shall be adhered to for operation involving nuclear safety. Contrary to the above, an operator did not follow in a timely manner the procedural steps described below subsequent to the feedwater transient that occurred during the December 8, 1971 incident.

Feedwater pump(s) was not shut down when the reactor water level exceeded 60 inches, as required by Reactor Operating Procedure 600-AN-I, Section I.

Feedwater regulator valve lockout was not reset (nor was there any attempt to do so) as required by Reactor Operating Procedure 600-AN-I, Section I.

The "Loss of Air to Feedwater Regulating Valve 'A'" and "Loss of Air to Feedwater Regulating Valve 'B'" annunciator stations flashed, and were acknowledged by an operator, but no corrective action was taken. This condition indicates a possible feedwater regulator valve "lockout" condition.

The condensate-booster pump tripped, and this condition was acknowledged; however, the operator did not take corrective action which could have minimized the magnitude of the reactor water level transient that followed.

The "Flow On" indicating light indicated a pump runout (110% feedwater flow) condition that was observed by the operator, but who did not take the corrective action which could have minimized the reactor water level transient.

The "Feedwater Pump Max Capacity" annunciator sounded, and was acknowledged by the operator, but with no apparent follow-up corrective action taken. This abnormal condition was a further indication of a feedwater pump runout condition.

B. Primary Containment

Section 6.1 of the facility Technical Specifications specifies that approved procedural steps covering a drywell pressurization (>2 psig) shall be adhered to for safe operation. Contrary to the above, the operator did not follow the procedural step described below, subsequent to

the drywell pressurization that occurred during the December 8, 1971 incident.

The containment (torus) sprays were not actuated when the pressure inside containment was greater than 2 psig, as required by Reactor Operating Procedure 1600-AN-I, Section E. The decision not to use the torus sprays was reached after it was inferred by calculation that the pressure inside the drywell had peaked at approximately 20 psig, and started to decrease. This occurred about 7 minutes after the reactor scram.

C. Procedures

The Station Review Board developed additional shutdown procedures for control of reactor water level and drywell pressurization subsequent to the incident, indicating that the procedures existing prior to the incident were deficient.

2.3 Control Panel Instrumentation and Controls

All instrumentation and controls were operable prior to the incident.

2.4 Plant Damage

The incident at the Dresden 3 facility on December 8, 1971, during which steam was released to the drywell, resulted in the following damage to the plant equipment and materials:

1. The "3A" electromatic valve solenoid operator was damaged by the steam jet from the "F" safety valve rendering the relief valve inoperable.

2. Most of the local power range monitor (LPRM) cables were damaged.

3. Sections of ventilating duct in the vicinity of the steam jet were dislodged.

4. One of seven containment cooling fan motors was found to have a moisture-caused ground.

5. Some damage to the piping insulation and the finish coat of drywell paint was caused by the steam jet.

As a consequence of the safety valve lift, it is estimated that 25,000 pounds of primary coolant were discharged to the drywell, resulting in a drywell pressure increase of about 20 psig. The drywell integrity was maintained and drywell pressure was reduced, due to pressure suppression system operation, to about 4 psig after 4 hours, about 1.75 psig after 12 hours, and drywell deinerting was instituted at about 40 hours.

Because of the primary coolant system activity at the time of the incident, fission product gases and particulates (I, Xe, Kr), as well as activation products were discharged to the drywell. Since drywell integrity was maintained, however, this activity was held up and decayed within the containment.

Drywell atmosphere samples indicated gross gamma activity levels 6×10^{-11} μCi/cc at about 40 hours. At 44 hours, after deinerting, initial entry for sampling was made and drywell inspection was initiated at 46 hours.

2.5 Corrective Actions

Prior to return to operation, all damaged components (see 2.4 above) were repaired or replaced and the reactor systems and components were returned to the conditions required by technical specifications.

Corrective actions were taken to reduce the probability and consequences of a similar event. The feedwater control valve air supply system was modified and tested to prevent valve lockout from insufficient air. An automatic feedwater pump trip at a 40-inch water level was installed on Dresden 2 and has been scheduled for installation on Dresden 3. Operating procedures were modified to clarify operator action when a high water level occurs and operator retraining was initiated. The direction of coolant discharge from the safety valve which actuated was changed to minimize damage from discharging steam, and the direction of discharge from other safety valves was verified to be acceptable. The LPRM cables were replaced with higher temperature rated cable on Dresden 2 during the recent refueling shutdown and identical cables have been obtained for later installation at Unit 3. The condensate-booster pump was

disassembled and inspected and all circuits were tested. A 60 psi drywell pressure recorder and a drywell temperature recorder were installed; however, the data collection has not been improved sufficiently to allow a complete analysis of a similar incident.

Following the December 8, 1971 incident, the licensee and the reactor vendor initiated an investigative program to determine the cause of premature lifting of safety valves. This program includes analytical and experimental investigations by a consultant of the effect of mechanical forces on safety valves. Additional data will be obtained by installing accelerometers on two safety valves and pressure transducers in two steam lines at the Quad Cities nuclear plant to obtain measurements during the start-up test program.(*) A pressure transducer was also installed in a Dresden 2 steam line. To eliminate uncertainties associated with setting safety valves with nitrogen, the licensee has ordered a steam test facility for testing safety valves. Both the licensee and the vendor have also initiated further studies to improve vessel water level control; however, the schedule for completing these programs was not made available to the study group.

The effectiveness of corrective actions taken to reduce the probability of occurrence of excessively high reactor vessel water levels could not be determined and has not been demonstrated. The corrective actions taken to date do not appear to be adequate to significantly reduce the probability of premature opening of safety valves. This conclusion is substantiated by the results of the Dresden 3 incident of May 4, 1972. Additional corrective actions planned for Dresden 3 and corrective actions resulting from the safety valve investigation program may assist in decreasing the probability and consequences of a steam discharge from a safety valve.

2.6 Documentation

The criteria used as the basis for evaluating acceptability of documentation is presented in Section 3.7 of this report. Following this incident, the licensee fulfilled all technical

(*) Letter, H. K. Hoyt to T. R. Wilson, June 3, 1972.

specification reporting requirements. The licensee's incident report of February 22, 1972, included a description of the incident, a damage assessment, operations assessment, a safety evaluation and a discussion of corrective actions. Additional information was provided by letter of April 26, 1972, in response to an AEC request. Both reports were received by the AEC subsequent to further operation of the plant.

The regulatory staff evaluation of the acceptability of the return to operation was based on the report submitted by the licensee on December 17, 1972, on site inspections totaling six days and reports by inspectors from the Division of Compliance and discussions between the licensee and the Division of Reactor Licensing. However, there is insufficient documentation in the regulatory docket files of the basis for the conclusion that return to operation was acceptable and that the licensee's analysis of the incident and the short term corrective actions were sufficient. There was no reason to doubt, however, that the licensee complied fully with the license and technical specifications and that the Division of Reactor Licensing was currently informed orally of the state of the plant, the licensee's conclusions and actions following the incident and his plan and schedule for resumption of operations. A final evaluation and staff conclusion concerning the long term investigations and corrective actions was deferred pending this report of the special study group.

As in the case of the Dresden 2 incident, documentation was insufficient to determine that the staff was satisfied with followup investigations and corrective actions at other facilities based on the experience at Dresden 3. Regulatory Operations, by telephone, questioned other licensees of operating Boiling Water Reactors about the potential for feedwater control valve lockout because of insufficient air. This followup indicated that a problem similar to the problem at Dresden 3 did not exist at other Boiling Water Reactors. This followup action was not documented in regulatory files. Lack of complete documentation is not intended to imply unsafe operation, inadequate review or erroneous conclusions. It is reported herein to assure that such experiences and precedents are fully understood by all licensees and reactor operators and by the regulatory staff and as a basis for recommending classification of the procedures and documentation.

2.7 Safety Significance

The significance of the radioactivity release during the incident at Dresden 3 was negligible with respect to the health and safety of the public and plant operating personnel. The activity level of radioactive gases released to the atmosphere was lower than the level allowed by Technical Specifications for release from the stack during normal plant operation.

The peak temperature in the drywell was measured to be 295°F and the peak pressure was about 20 psig. Although the design temperature and pressure are 62 psig and 280°F, the licensee, in their Special Report of Incident of June 5, 1970, has shown that the incident conditions were less severe than the design conditions.

The December 8 incident did not incapacitate any systems important to the safety of the plant. However, because of the high temperature, pressure and humidity caused by the discharge of steam and water into the drywell, neutron monitoring (LPRM) cables and electromatic relief valve position switches were damaged sufficient to cause improper information to be provided to the control room. Other means were available to check the reactivity status of the reactor and the status of these valves.

The Dresden 3 incident of December 8, 1971, did not result in the release of significant radioactivity to the environment and the key plant systems and components were available to cope with the reactor water level transient and the failure of the isolation condenser to operate without endangering the fuel cladding.

3.0 Dresden Unit 3 Incident, May 4, 1972

This incident was initiated while the reactor was operating at 100% power at near equilibrium conditions; the reactor scrammed for reasons unknown.

3.1 Performance of Plant Systems and Components

3.1.1 Feedwater System

3.1.1.1 System Requirements

Refer to System Requirements contained in Section 1.1.1 of Appendix C for the Dresden 2 June 5, 1970 incident.

3.1.1.2 System Performance

The reactor vessel water level was at +28" prior to the incident. Two feedwater pumps were operating with the third in standby. The feedwater flow rate was 9 x 10^6 lbs/hr. Following the scram, the water level dropped to approximately -1". The two operating feed pumps increased their flow rate to 10.6 x 10^6 lbs/hr. The pumps did not go into the flow control mode. When the level increased to approximately +28", the operator tripped both feedwater pumps. The reactor vessel water level continued to rise and reached a maximum level of about +54". This level is approximately 4" below the HPCI and the isolation condenser steam inlet pipes and at least 4 feet below the main steam lines. The feedwater control valve did not experience a lockout as it did during the incident of December 8, 1971.

3.1.1.3 Evaluation of System Performance

The initial decrease in water level of approximately 30" is the expected result of the scram and void collapse. It appears that the feedwater control system would not have automatically maintained water level within ± 30" based upon the observed operator intervention and the maximum water level reached during the incident.

3.1.2 Overpressure Protection (Relief and Safety Valves)

3.1.2.1 System Requirements

Refer to System Requirements contained in Section 1.1.2 of Appendix C for the Dresden 2 June 5, 1970 incident.

3.1.2.2 System Performance

Following the spurious scram, reactor system pressure decreased to 828 psig (1 minute) and then began to increase. In accordance with design. The closure of main steam isolation valves did not occur at 850 psig because the reactor mode switch had been taken out of the "run" mode. However, the main steam isolation valves did close automatically as intended after approximately 5 minutes when a reactor vessel water level of +48" was reached. Reactor pressure then increased to 1060 psig (10 minutes) and an unsuccessful attempt was made by the operator to place the isolation condenser in service. Pressure continued to increase to 1110 psig (12 minutes) at which time the "A" relief valve operated automatically for approximately 15 seconds and reduced pressure to 1040 psig. At approximately the time that the relief valve operated, the "A" safety valve opened momentarily and prematurely. Other pressure increases were experienced during the transient and controlled by manual operation of the "B" relief valve to maintain pressure less than 1100 psig.

The isolation condenser was placed in service after approximately 25 minutes to reduce reactor pressure in an orderly manner.

3.1.2.3 Evaluation of System Performance

During the incident, one relief valve was actuated 27 psi below its required setting of 1135 psig. The safety valve, whose set pressure was 100 psi above the highest recorded system pressure reached during the transient, should not have opened. From all available data, the safety valve remained open only for a few seconds.

The initiating mechanism for premature opening of the safety valve during this incident may have been the operation of the adjacent relief valve. This actuating mechanism has not been confirmed.

3.1.3 Containment System

3.1.3.1 System Requirements

Refer to System Requirements contained in Section 1.1.3 in Appendix C for the Dresden 2 June 5, 1970 incident.

3.1.3.2 System Performance

During the incident the main steam line isolation valves closed on "high" reactor water level (48"). Following the opening of the safety valves, the containment isolated (Group II) on high drywell pressure (2 psig.). The primary containment was pressurized to 2.5 psi and subjected to a temperatue of 180°F. Drywell purge and deinerting were initiated 12 hours after the reactor scram. Containment integrity was maintained until sampling indicated purging was permissible.

3.1.3.3 Evaluation of System Performance

The containment isolation system functioned as designed. Containment pressure and temperatures that were experienced were within the design values. Containment response appears to have been normal.

The main steam isolation valves closed at a high reactor vessel water level of +48" as a result of recent installation of a trip circuit that functions whenever the mode switch is in other than the "run" mode. This trip function was not specified in the Dresden 3 technical specifications.

3.1.4 Reactivity Control Systems

3.1.4.1 System Requirements

Refer to System Requirements contained in Section 1.1.4 of Appendix C for the Dresden 2 June 5, 1970 incident.

3.1.4.2 System Performance

The reactor scrammed from spurious signals which may have originated in the reactor vessel low water level sensor. The shutdown status of the reactor was verified immediately.

3.1.4.3 Evaluation of Performance

The computer indicated a half scram condition on one of the two low water level sensors. There were no other indications of abnormal conditions. The exact cause of the scram is unknown. The reactor protective system computer input relays were examined and adjusted.

3.1.5 Electrical Systems

3.1.5.1 System Requirements

Refer to System Requirements contained in Section 1.1.5 of Appendix C for the Dresden 2 June 5, 1970 incident.

3.1.5.2 System Performance

No documentation was available to determine response of the auxiliary power system during incident. Both diesel generators started automatically upon receipt of the 2 psig high drywell pressure signal. The diesel generators continued to operate until the high drywell pressure signal cleared approximately 23 minutes after reactor scram.

3.1.5.3 Evaluation of Performance

It appears that the electrical system performed as designed.

3.1.6 Emergency Core Cooling Systems

3.1.6.1 System Requirements

Refer to System Requirements contained in Section 1.1.6 of Appendix C for the Dresden 2 June 5, 1970 incident.

3.1.6.2 System Performance

Approximately 12-1/2 minutes after the scram, a 2 psig high drywell pressure signal was received. As designed, this signal caused automatic start of all ECCS pumps with the exception of the HPCI. The HPCI received an initiating signal, but failed to start due to the presence of a turbine trip signal from high water level (+48") in the reactor vessel. Based on the available information, the maximum water level experienced during the transient (+55") remained below the elevation of the nozzle that supplies steam to the HPCI (≈60"). Injection of water into the reactor vessel from any of the emergency core cooling systems was not required and did not occur. The automatic pressure relief system was not initiated as the necessary and required conditions were not both present. The ECCS systems that were initiated automatically were not secured until the high drywell pressure condition had cleared (40 minutes).

3.1.6.3 Evaluation of Performance

Actuation of emergency core cooling systems was initiated and they performed in accordance with design.

The HPCI was not required to inject during the transient as coolant inventory was well in excess of that required. The HPCI steam supply line did not flood because the reactor water level transient was minimized by operator action in controlling the feedwater pumps manually. The HPCI would have been available for auto restart if water level had desreased to 45".

3.1.7 Isolation Condenser

3.1.7.1 System Requirements

Refer to System Requirements contained in Section 1.1.7 in Appendix C for the Dresden 2 June 5, 1970 incident.

3.1.7.2 System Performance

The condenser return valve failed to open when the operator attempted manually to place the isolation condenser in service. An operator was dispatched to crank the valve off its seat. The valve then operated remotely and was effective in controlling the pressure late in the transient.

3.1.7.3 Evaluation of Performance

The system was unable to perform its intended functions due to valve failure.

3.1.8 Reactor Coolant Cleanup System

3.1.8.1 System Requirements

Refer to System Requirements contained in Section 1.1.8 in Appendix C for the Dresden 2 June 5, 1970 event.

3.1.8.2 System Performance

There was no documentation available to determine the response of the Reactor Coolant Cleanup System during the transient.

Blowdown of primary system water was established through the system to the main condenser 40 minutes after the incident began.

3.1.8.3 Evaluation of Performance

The reactor water level transient experienced was sufficent to automatically isolate the system. No documentation was available to determine actual system performance.

3.2 Control Room Operations

3.2.1 Qualifications of Operating Personnel

Control room operating personnel on duty at the time of the incident were as follows:

Title	Type of License
Shift Engineer	SRO
Shift Foreman	SRO
Control Operator	RO
Center Desk Operator	RO
Sontrol Operator (Unit #1)	RO
Control Operator (Unit #2)	RO

Additional day shift licensed operating personnel who were available during the incident included: One Shift Engineer, one Shift Foreman, and several reactor operators.

All of the licensed operating personnel involved during the incident of May 4, 1972, had previous operating experience on Dresden Unit 1 and 2. The duty Shift Foreman, who is a graduate engineer, received his senior operator's license in Febraury 1971. The duty Control Operator had been licensed for Dresden Units 2 and 3 in February 1970, at the time of the Unit 3 incident. The Center Desk Operator received his operating license for Dresden Units 1 and 2 in November 1967 and August 1969, respectively.

Subsequent to the May 4, 1972 incident, and as part of the scheduled retraining program, Operations supervision reviewed with all licensed operators, the latest operating and emergency procedures for: (1) excessively high reactor water level, and (2) drywell pressurization. Moveover, a simulator training refresher course was initiated (on a rotational basis) in January 1972 for all license reactor operating personnel. To date, approximately two-thirds (24 operators) of the reactor operators have received this portion of the retraining program.

3.2.2 Operator Actions

3.2.2.1 Feedwater Control System

In accordance with Section 6.2.A of the facility Technical Specifications, there were approved written procedures covering the feedwater and reactor water level transients that were adhered to by operating personnel during the incident. It should be noted, however, that the water level at which the operating personnel are instructed to trip the reactor feedwater pump(s) was lowered from 48-inches to 25 to 30 inches following this incident.

3.2.2.2 Primary Containment

Operations personnel did follow the required procedures subsequent to the drywell pressurization that occurred during the May 4, 1972 incident.

The containment (torus) sprays were actuated when the pressure inside containment was greater than 2 psig, as required by the Reactor Operating Procedure 1600-AN-I, Section E.

3.2.2.3 Primary Coolant Pressure Relief System

Even though the correct operator action was taken regarding a high reactor pressure condition, no detailed written procedures were available for the operator to refer to at the time of the May 4, 1972 incident. To date, only minimal instructions for the operating personnel regarding high pressure conditions are found in 900-AN-I, Annunciator Procedures (ANN-58).

3.2.2.4 Reactor Water Level

Operations acknowledged the "High Reactor Water Level" alarm, and did respond to correct the condition.

3.2.2.5 Procedures

On the basis of the operator performance during the May 4, 1972 blowdown incident, it appeared that the operating procedures had become more adequate to cope with an incident resulting from reactor feedwater and water level transients.

3.3 Control Panel Instrumentation and Controls

All instrumentation and controls were operable prior to the incident.

3.4 Plant Damage

The incident at the Dresden 3 facility on May 4, 1972, during which steam was released to the torus by automatic actuation of the electromatic pressure relief valves and to the drywell by faulty actuation of a safety valve, resulted in only minor plant damage. Steam from a safety valve struck the line leading from electromatic relief valve (3A) to the torus and the base of electromatic relief valve (3E). The impingement of steam on the base of valve 3E tore a piece of insulation loose. Other structures and components within the drywell were inspected and found to be undamaged by the release of steam.

The primary coolant discharge to the drywell resulted in a drywell pressure increase of about 2-1/2 psig. The drywell pressure was reduced to normal within 30 minutes of the coolant discharge. Because of the primary coolant system activity at the time of the incident, fission product gases and particulates (I, Xe, Kr), as well as activation product gases and particulates, were discharged to the drywell. Since drywell integrity was maintained during the incident, this activity was held up and permitted to decay. Approximately one hour and eleven minutes following the discharge to the drywell, the drywell atmosphere was sampled and found to be within specifications for venting. The specification limits for drywell venting are that the gross activity shall not exceed the facility stack release limit.

3.5 Corrective Actions

Prior to return to operation the damaged components were repaired or replaced and the reactor returned to conditions required by technical specifications. Although these actions restored the affected systems to the "as designed" conditions, they do not preclude recurrence of similar transients. The cause of the spurious scram was not determined and, therefore, the potential exists for additional scrams of an identical nature. The isolation condenser valve which failed was serviced and made operable; however, no corrective action was taken to improve the reliability of the equipment required to place the emergency condenser in service. In addition, the performance of the feedwater control system following a scram still resulted in reactor vessel water levels greater than the Group I isolation initiation setpoint of 48".

The ongoing programs implemented by the licensee after the December 8, 1971 incident are continuing.

3.6 Documentation

Following this incident, the licensee fulfilled the technical specification reporting requirements. The licensee's incident report of June 3, 1972, includes a description of the incident, damage and operations assessments and a discussion of corrective actions. This report was received subsequent to return to operation. The staff evaluation of the acceptability of return to operation was based on a TWX from the licensee and an on-site inspection by Regulatory Operations inspectors. However, there is no documentation in the Regulatory docket files of the basis for the conclusion that return to operation was acceptable and that the licensee's analysis of the incident and the short term corrective actions were sufficient. The licensee, of course, is required to comply with his license and technical specifications in any event.

3.7 Safety Significance

The direct safety significance of the May 4, 1972, incident at Dresden 3 was negligible with respect to the health and safety

of the public and plant operating personnel. The activity level of radioactive gases vented to the atmosphere was lower than the level allowed by Technical Specifications for release from the stack during normal plant operation.

The measured drywell pressure and temperature resulting from the incident were 2.5 psig and 180°F, respectively. The design capability of the drywell containment is 62 psig and 280°F.

Since the May 4 incident did not result in the release of significant radioactivity to the environs and since systems and components were available to cope with the transient and the failure of the isolation condenser to operate without endangering the fuel cladding or the primary system boundary, we conclude that this incident had minimal safety significance to the public.

APPENDIX D

RADIOLOGICAL EFFECTS

The results of radiological measurements made at the time of the incident were evaluated to determine the radiological consequences of the coolant release incidents on the health and safety of plant operating personnel and the public. These data consisted of the radiation doses received by plant operating personnel during the period of interest, as determined from film badge readings, the radioactivity release rate through the plant stack following the incident and where available, the results of analyses of air, soil and grass samples taken from outside the plant boundaries. These data were collected by licensee personnel and were reviewed by AEC field inspectors in the course of their investigation.

The maximum radiation dose received by any individual onsite during the period covering the incidents are summarized in Table I.

TABLE I

Plant Name	Date of Incident	Maximum Dose to Any Individual	Duration at Film Badge Period
Dresden	6/5/70	570 Mrem	10 days
Monticello	9/28/71	310 Mrem	2 weeks
Millstone-1	10/10/71	530 Mrem	1 month
Nine Mile Point	12/31/71	230 Mrem	2 weeks
Dresden-3	12/8/72	780 Mrem[1/]	1 week
Monticello	2/26/72	410 Mrem	2 weeks
Nine Mile Point	2/28/72	480 Mrem	2 weeks
Dresden-3	5/4/72	820 Mrem	2 weeks

The name of the nuclear plant, date of the incident, maximum film badge reading of any individual and film badge period are shown in Table I. It should be emphasized that the radiation doses shown are the maximum total dose received by any individual in performing assigned duties during the period and, therefore, were not necessarily received as a result of the incident. For example, in the case of Dresden 3, the majority of the radiation exposure was obtained in the course of performing duties associated with the plant in-service inspection program at Dresden Unit-1. None of the doses received by plant personnel exceeded the quarterly dose permitted under 10 CFR Part 20.

1/ Exposure resulted from in-service inspection of Dresden Unit-1.

The stack release rates prior to and during the 24-hour period following the incidents are shown in Table II.

The maximum stack release limits permitted by the technical specifications for a given plant are dependent upon many factors such as stack height, site meteorology and size of the site and vary from about 270,000 to 800,000 microcuries per second. As may be seen, the release rates were far below permissible limits.

Air, grass, and soil samples were taken outside the Dresden 2 and 3 plant boundaries by the licensee following the incidents of June 5, 1970, and December 8, 1971. The grass and soil samples were analyzed for radioactivity by independent outside laboratories. None of these environmental samples revealed that any measurable quantities of radioactivity were released to the environment.

TABLE II

		Stack Release Rate Prior to Incident	Maximum Release Rate During 24 hr. Period Following the Incident	Percentage of the Tech. Spec. Annual Average Limit
Dresden-2	6/5/70	25,000	25,000	4%[1/]
Monticello-1	9/28/71	10,000	45,500	17%[2/]
Millstone-1	10/10/71	46,000	No increase	-
Nine Mile Point-1	12/21/71	21,000	33,500	4%[3/]
Dresden-3	2/8/72	25,000	No increase	-
Monticello-1	2/26/72	10,000	54,000	20%[2/]
Nine Mile Point-1	2/28/72	19,200	No increase	-
Dresden-3	5/4/72	6,900	52,000[(4)]	8%[1/]

1/ The technical Specifications require that the gaseous radioactive effluents from the combined Dresden 1, 2, and 3 stations be maintained at less than a yearly average limit of ∿900,000 μCi/sec.

2/ The Technical Specifications require that the gaseous radioactive effluents from the Monticello-1 facility be maintained at less than a yearly average limit of ∿270,000 μCi/sec.

3/ The Technical Specifications require that the gaseous radioactive effluents from Nine Mile Point-1 be maintained at less than a yearly average limit of ∿800,000 μCi/sec.

(4) Due to starting of mechanical vacuum pump on the main condenser.

APPENDIX E

LIST OF REFERENCES

1. USAEC memo, H. D Thornburg to J. P. O'Reilly, 6/10/70, Dresden 2.

2. USAEC Inspection Report 237/70-8, Dresden 2.

3. USAEC trip report, V. Stello to R. S. Boyd, 7/1/70, Dresden 2.

4. USAEC memo, R. H. Engelken to P. A. Morris, 6/11/70, Dresden 2.

5. Commonwealth Edison Company letter, B. Lee to P. A. Morris, 6/12/70, Dresden 2.

6. USAEC memo, B. H. Grier to L. D. Low, 6/12/70, Dresden 2.

7. Commonwealth Edison Company letter, B. Lee to P. A. Morris, 6/15/70, Dresden 2.

8. USAEC memo, H. D. Thornburg to J. P. O'Reilly, 7/2/70, Dresden 2.

9. Commonwealth Edison Company letter, B. Lee to P. A. Morris, 7/6/70, Dresden 2.

10. USAEC memo, R. H. Engelken to P. A. Morris, 7/6/70, Dresden 2.

11. USAEC Inspection Report 237/70-9, Dresden 2.

12. USAEC letter, P. A. Morris to B. Lee, 7/17/70, Dresden 2.

13. USAEC letter, P. A. Morris to B. Lee, 7/24/70, Dresden 2.

14. Commonwealth Edison Company letter, B. Lee to P. A. Morris, 7/27/70, Dresden 2.

15. USAEC memo, P. F. Collins to D. J. Skovholt, 8/3/70, Dresden 2.

16. USAEC memo, D. R. Roth to P. F. Collins, 8/3/70, Dresden 2.

17. USAEC memo, H. D. Thornburg to J. P. O'Reilly, 8/28/70, Dresden 2.

18. USAEC Inspection Report 237/70-10, Dresden 2.

19. USAEC memo, H. D. Thornburg to J. P. O'Reilly, 10/8/70, Dresden 2.

20. USAEC memo, R. H. Engelken to R. S. Boyd, 9/30/70, Dresden 2.

21. USAEC memo, R. H. Engleken to R. S. Boyd, 11/9/70, Dresden 2.

22. Commonwealth Edison Company Safety Analysis Report, Volumes I-VI, 11/17/67, Dresden 2.

23. USAEC Inspection Report 249/71-10, Dresden 3.

24. USAEC Notification of an Incident or Occurrence #48, 12/9/71, Dresden 3.

25. Commonwealth Edison Company letter to P. A. Morris, 12/1/71, Dresden 3.

26. Commonwealth Edison Company letter, B. Lee to P. A. Morris, 2/22/72, Dresden 3.

27. USAEC memo, P. A. Morris to E. G. Case, 3/22/72, Dresden 3.

28. USAEC letter, D. J. Skovholt to B. Lee, 3/29/72, Dresden 3.

29. USAEC memo, R. H. Engelken to R. S. Boyd and D. J. Skovholt, 4/6/72, Dresden 3.

30. USAEC memo, D. Knuth to D. J. Skovholt, 4/20/72, Dresden 3.

31. Commonwealth Edison Company letter, B. Lee to D. J. Skovholt, 4/26/72, Dresden 3.

32. USAEC Notification of an Incident or Occurrence #54, 5/5/72, Dresden 3.

33. USAEC Inspection Report 249/72-2, Dresden 3.

34. Commonwealth Edison Company TWX to B. H. Grier, 5/6/72, Dresden 3.

35. USAEC letter, B. H. Grier to B. Lee, 5/8/72, Dresden 3.

36. Commonwealth Edison Company TWX, W. P. Worden to P. A. Morris, 5/8/72, Dresden 3.

37. USAEC memo, G. Fiorelli to J. G. Keppler, 5/10/72, Dresden 3.

38. USAEC letter, B. H. Grier to B. Lee, 5/12/72, Dresden 3.

39. Commonwealth Edison Company letter, W. P. Worden to P. A. Morris, 5/12/72, Dresden 3.

40. USAEC memo, R. H. Engelken to D. J. Skovholt, 5/18/72, Dresden 3.

41. Commonwealth Edison Company letter, W. P. Worden to E. J. Bloch, 6/3/72, Dresden 3.

42. Commonwealth Edison Company Safety Analysis Report, Volumes I-VI, 11/17/67.

43. USAEC Inspection Report 263/71-13, Monticello.

44. USAEC Inspection Report 263/71-14, Monticello.

45. Northern States Power Company letter, R. O Duncanson to P. A. Morris, 10/5/71, Monticello.

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