NUREG-0823

Integrated Plant Safety Assessment
Systematic Evaluation Program

Dresden Nuclear Power Station, Unit 2

Commonwealth Edison Company
Docket No. 50-237

Final Report

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

February 1983

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ABSTRACT

The Nuclear Regulatory Commission (NRC) has published its Final Integrated Plant Safety Assessment Report (IPSAR) (NUREG-0823), under the scope of the Systematic Evaluation Program (SEP), for Commonwealth Edison Company's Dresden Nuclear Power Station, Unit 2, located in Grundy County, Illinois. The SEP was initiated by the NRC to review the design of older operating nuclear reactor plants to reconfirm and document their safety. This report documents the review completed under the SEP for Dresden Unit 2. The review has provided for (1) an assessment of the significance of differences between current technical positions on selected safety issues and those that existed when Dresden Unit 2 was licensed, (2) a basis for deciding on how these differences should be resolved in an integrated plant review, and (3) a documented evaluation of plant safety when all supplements to the Final IPSAR and the Safety Evaluation Report for converting the license from a provisional to a full-term license have been issued. The report also addresses the comments and recommendations made by the Advisory Committee on Reactor Safeguards in connection with its review of the Draft Report, issued in October 1982. The Final IPSAR and its supplements will form part of the bases for considering the conversion of the existing provisional operating license to a full-term operating license.

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ACRONYMS AND INITIALISMS

APRM	average power range monitor
ASME	American Society of Mechanical Engineers
BTP	Branch Technical Position
BWR	boiling-water reactor
CECo	Commonwealth Edison Company
CFR	<u>Code of Federal Regulations</u>
CST	condensate storage tank
DBE	design-basis event
DER	design electrical rating
DG	diesel generator
ECCS	emergency core cooling system
EHC	electrohydraulic control
EI&C	electrical instrumentation and control
FSAR	Final Safety Analysis Report
FTOL	full-term operating license
FWCI	feedwater coolant injection
GDC	General Design Criterion(a)
GE	General Electric Company
gpm	gallons per minute
HEPB	high energy pipe break
hp	horsepower
HPCI	high-pressure coolant injection
IE	Office of Inspection and Enforcement
IEEE	Institute of Electrical and Electronics Engineers
IPSAR	Integrated Plant Safety Assessment Report
IREP	Integrated Reliability Evaluation Program
IRM	intermediate range monitor
LCO	limiting condition for operation
LER	licensee event report
LOCA	loss-of-coolant accident
LPCI	low-pressure coolant injection
LPRM	local power range monitor
LWR	light-water reactor
MCC	motor control center
MCPR	minimum critical power ratio
MDC	maximum dependable capacity
MOV	motor-operated valve
mph	miles per hour
MSIV	main steam isolation valve
MSL	mean sea level
MWe	megawatt-electric
MWt	megawatt-thermal
NRC	U.S. Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PMF	probable maximum flood
PMP	probable maximum precipitation
POL	provisional operating license

PRA	probabilistic risk assessment
psi	pounds per square inch
psig	pounds per square inch gage
PWR	pressurized-water reactor
RBCCW	reactor building closed cooling water
RCPB	reactor coolant pressure boundary
RPS	reactor protection system
RSCS	reactor shutdown cooling system
RWCU	reactor water cleanup
SALP	Systematic Appraisal of Licensee Performance
SAR	safety analysis report
SBGTS	standby gas treatment system
SEP	Systematic Evaluation Program
SER	safety evaluation report
SRP	Standard Review Plan
STS	Standard Technical Specification
SWS	service water system
TMI	Three Mile Island
UHS	ultimate heat sink
USI	unresolved safety issue

SUMMARY

The Systematic Evaluation Program (SEP) was initiated by the U.S. Nuclear Regulatory Commission (NRC) to review the designs of older operating nuclear reactor plants to reconfirm and document their safety. The review provides (1) an assessment of the significance of differences between current technical positions on safety issues and those that existed when a particular plant was licensed, (2) a basis for deciding on how these differences should be resolved in an integrated plant review, and (3) a documented evaluation of plant safety.

The review compared the as-built design with current review criteria in 137 different areas defined as "topics." The "Definition" and other information for each of these topics appear in Appendix A. During the review, 49 of the topics were deleted from consideration by the SEP because a review was being made under other programs (Unresolved Safety Issue (USI) or Three Mile Island (TMI) Action Plan Tasks), or the topic was not applicable to the plant; that is, the topic was applicable to pressurized-water reactors rather than to boiling-water reactors. The topics deleted because they were being reviewed under either the USI or TMI programs are listed in Appendix B, and the topics deleted because they did not apply to the plant are listed in Appendix C. The status of the USI and TMI tasks will be addressed in a provisional operating license conversion safety evaluation report. That report will be issued following completion of the SEP Integrated Plant Safety Assessment Report (IPSAR) and together with the IPSAR will be available for considering the conversion of the Dresden Unit 2 provisional operating license to a full-term operating license.

Of the original 137 topics, 88 were, therefore, reviewed for Dresden Unit 2; of these, 54 met current criteria or were acceptable on another defined basis. No modifications were made by the licensee during topic review. References for correspondence pertaining to safety evaluation reports (SERs) for each of the 88 topics appear in Appendix E.

The review of the remaining 34 topics found that certain aspects of plant design differed from current criteria. The topics that differed from current licensing criteria consisted of 73 individual issues. These issues were considered in the integrated assessment of the plant, which consisted of evaluating the safety significance and other factors of the identified differences from current design criteria to arrive at decisions on whether backfitting was necessary from an overall plant safety viewpoint. To arrive at these decisions, engineering judgment was used as well as the results of a limited probabilistic risk assessment study. This study and staff comments are in Appendix D.

Table 4.1 summarizes the staff's backfitting positions reached in the integrated assessment. In general, backfit requirements fell into one or more of the following categories: (1) equipment modification or addition, (2) procedure development or Technical Specification changes, (3) refined engineering analysis or continuation of ongoing evaluation, and (4) no backfit modifications necessary. Eight issues primarily require equipment modification or addition;

seventeen issues primarily require procedure development or changes; and twenty-three issues primarily require refined engineering analysis or continuation of an ongoing evaluation. Twenty-five issues do not require any backfitting.

Safety improvements are being planned as a result of the integrated assessment and are listed below. Some safety improvements have already been implemented by the licensee. The following descriptions summarize the backfit actions addressed by the integrated assessment. The sections in this report relating to the issue are given in parentheses.

SAFETY IMPROVEMENTS AGREED TO AND TO BE IMPLEMENTED BY THE LICENSEE AS A RESULT OF SEP

These improvements fall into three categories. The first category comprises hardware modifications or additions that the licensee has agreed to make and that are required by the NRC. The second category comprises procedural or Technical Specification changes that become part of the operating license. The third category comprises additional engineering analysis followed by corrective measures where required. These three categories are listed below, and the issues are discussed in sections of this report given in parentheses.

Category 1, Equipment Modifications or Additions Required by NRC

(1) Modify roof parapets to ensure ponded water is within roof load capacity (4.1.3).

(2) Provide locking devices for manual isolation valves (4.18.3).

(3) Provide second isolation valve on containment penetration branch lines (4.18.6).

(4) Modify existing dc power system monitoring for breaker or fuse position and battery availability (4.23.3 and 4.28).

(5) Install Class 1E protection at interface of reactor protection system and its power supply (4.24.3).

(6) Modify diesel generator annunicators (4.26.1).

(7) Provide for bypassing the diesel generator underfrequency protective trip during accident conditions (4.26.2).

Category 2, Technical Specification Changes and Procedure Development

The staff's position regarding Technical Specification changes is that the proposed Technical Specification changes may be submitted all together following the completion of the integrated assessment. The licensee should submit within 90 days after the issuance of the Final Integrated Plant Safety Assessment Report a request for an amendment of the operating license to change the facility Technical Specifications.

(1) Modify existing flood emergency plan to provide ability to cope with design-basis flood (4.1.2 and 4.1.4).

(2) Modify water control structures inspection program to ensure it is overseen by qualified personnel and that special inspections are conducted following extreme events (4.4.3).

(3) Develop procedures for achieving cold shutdown from outside the control room (4.15 and 4.25.1).

(4) Provide procedures for testing shutdown cooling system temperature interlocks (4.17 and 4.25.4).

(5) Provide mechanical locking devices and administrative procedures to ensure valve closure (4.18.1).

(6) Modify procedures for postaccident engineered safety features leakage (4.18.2).

(7) Provide procedures to ensure disconnect links between redundant electrical divisions are open (4.21.2).

(8) Provide assurance that tie breakers are not used during power operations (4.21.3).

(9) Limit allowable time for obtaining diesel generator DG 2/3 control power from Unit 3 (4.21.4).

(10) Prohibit paralleling of shared dc systems during power operations (4.23.1).

(11) Prohibit placing DG 2/3 switch in "bypass" during normal operation (4.23.2).

(12) Revise procedures to achieve cold shutdown using safety-grade systems (4.25.2).

(13) Modify plant Technical Specification limits for primary coolant system iodine activity (4.31 and 4.32).

Category 3, Additional Engineering Evaluation

It is the staff's position regarding additional engineering evaluation that all evaluations and corresponding backfits and schedule for backfit implementations be submitted within the established schedules, as documented in the appropriate report sections and summarized in Table 4.1. These evaluations are as follows:

(1) Identify radiography requirements of vessels and pump casing (4.2.1).

(2) Demonstrate fracture toughness for various components or that failure consequence is acceptable (4.2.2).

(3) Ensure failure of ventilation stack will not affect safe shutdown (4.3.2).

(4) Identify and ensure components outside qualified structures can withstand tornado loading or their loss will not affect safe shutdown (4.3.3).

(5) Demonstrate failure of roof decks will not affect plant safety (4.3.4).

(6) Demonstrate structural capability of plant to withstand load combinations (4.3.5 and 4.10)

(7) Ensure operability of DG 2 and DG 2/3 following loss of ventilation systems resulting from tornado missiles (4.5.3).

(8) Ensure capability to achieve safe shutdown using tornado-missile-protected systems (4.5.4).

(9) Provide schedule and basis for reinspection of low-pressure turbines (4.6).

(10) Address effects of jet impingment on target pipe (4.7.1).

(11) Demonstrate deformation of pipe associated with global strain will not affect functionability (4.7.2).

(12) Ensure detectability for through-wall cracks in high-energy fluid systems piping (4.7.3).

(13) Provide criteria and results of pipe whip load formulation and ensure pipe whip and jet impingement will not affect containment liner (4.7.4).

(14) Determine seismic capability of mechanical equipment (4.9.2).

(15) Provide analysis of structural integrity of cable trays (4.9.3).

(16) Ensure adequate setpoints for thermal overload protection of motor-operated valves or bypass thermal overloads (4.12.1).

(17) Provide leakage detection capability in conjunction with pipe breaks inside containment (4.13.1).

(18) Provide seismically qualified leakage detection system (4.13.2).

(19) Ensure adequacy of protective relaying (4.21.1).

(20) Demonstrate adequate isolation of Class 1E sources from non-Class 1E loads (4.21.5).

(21) Ensure common-mode electrical faults will not disable the neutron flux monitoring systems (4.24.1).

(22) Ensure the reactor protection system is protected from faults generated in process computer (4.24.2).

TOPIC SAFETY EVALUATION REPORTS

Copies of this report and the associated safety evaluation reports for the 88 topics listed in Appendix E are available for public inspection at the NRC Public Document Room, 1717 H Street N.W., Washington, D.C. 20555 and at the Morris Public Library, 604 Liberty Street, Morris, Illinois 60451. Copies of this report are also available for purchase from sources indicated on the inside front cover.

The review of the 88 topics was performed by the NRC staff and contractors listed in Appendix G. The Integrated Assessment Team performing the integrated assessment on the 34 topics that did not meet current criteria is as follows:

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INTEGRATED PLANT SAFETY ASSESSMENT
SYSTEMATIC EVALUATION PROGRAM
DRESDEN NUCLEAR GENERATING STATION, UNIT 2

1 INTRODUCTION

1.1 Background

In the late 1960s and early 1970s, the U.S. Atomic Energy Commission's (now Nuclear Regulatory Commission) scope of review of proposed power reactor designs was evolving and somewhat less defined than it is today. The requirements for acceptability evolved as new facilities were reviewed. In 1967, the Commission published for comment and interim use proposed General Design Criteria for Nuclear Power Plants (GDC) that established minimum requirements for the principal design standards. The GDC were formally adopted, though somewhat modified, in 1971, and have been used as guidance in reviewing new plant applications since then. Safety guides issued in 1970 became part of the Regulatory Guide Series in 1972. These guides describe methods acceptable to the staff for implementing specific portions of the regulations, including certain GDC, and formalize staff techniques for performing a facility review. In 1972, the Commission distributed for information and comment a proposed "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," now Regulatory Guide 1.70. It provided a standard format for these reports and identified the principal information needed by the staff for its review. The Standard Review Plan (SRP, NUREG-75/087) was published in December 1975 and updated in July 1981 (NUREG-0800) to provide further guidance for improving the quality and uniformity of staff reviews, to enhance communication and understanding of the review process by interested members of the public and nuclear power industry, and to stabilize the licensing process. For the most part, the detailed acceptance criteria prescribed in the SRP are not new; rather they are methods of review that, in many cases, were not previously published in any regulatory document.

Because of the evolutionary nature of the licensing requirements discussed above and the developments in technology over the years, operating nuclear power plants embody a broad spectrum of design features and requirements depending on when the plant was constructed, who was the manufacturer, and when the plant was licensed for operation. The amount of documentation that defines these safety-design characteristics also has changed with the age of the plant--the older the plant, the less documentation and potentially the greater the difference from current licensing criteria.

Although the earlier safety evaluations of operating facilities did not address many of the topics discussed in current safety evaluations, all operating facilities have been reviewed more recently against a substantial number of major safety issues that have evolved since the operating license was issued. Conclusions of overall adequacy with respect to these major issues (e.g., emergency core cooling system, fuel design, and pressure vessel design) are a matter of record. On the other hand, a number of other issues (e.g., seismic considerations, tornado and turbine missiles, flood protection, pipe break effects inside containment, and piping whip) have not been reviewed against today's acceptance criteria for many operating plants, and documentation for them is incomplete.

1.2 Systematic Evaluation Program Objectives

The Systematic Evaluation Program (SEP) was initiated by the U.S. Nuclear Regulatory Commission (NRC) in 1977 to review the designs of older operating nuclear reactor plants in order to reconfirm and document their safety. The review provides (1) an assessment of the significance of differences between current technical positions on safety issues and those that existed when a particular plant was licensed, (2) a basis for deciding on how these differences should be resolved in an integrated plant review, and (3) a documented evaluation of plant safety.

The original SEP objectives were:

(1) The program should establish documentation that shows how the criteria for each operating plant reviewed compare with current criteria on significant safety issues, and should provide a rationale for acceptable departures from these criteria.

(2) The program should provide the capability to make integrated and balanced decisions with respect to any required backfitting.

(3) The program should be structured for early identification and resolution of any significant deficiencies.

(4) The program should assess the safety adequacy of the design and operation of currently licensed nuclear power plants.

(5) The program should use available resources efficiently and minimize requirements for additional resources by NRC or industry.

The program objectives were later interpreted to ensure that the SEP also provides safety assessments adequate for conversion of provisional operating licenses (POLs) to full-term operating licenses (FTOLs). The final version of this report and a POL conversion safety evaluation report that will address the status of all applicable generic activities (TMI and USI), including those that formed the basis for deletion of specific SEP topics, will form a part of the basis for the Commission's consideration of the license conversion.

Many of the plants selected for review were licensed before a comprehensive set of licensing criteria had been developed. They include five of the oldest nuclear reactor plants and seven plants under NRC review for the conversion of POLs to FTOLs. The plants to be considered under the original Phase II program were

(1) Yankee Rowe (FTOL PWR)
(2) Haddam Neck (FTOL PWR)
(3) Millstone 1 (POL BWR)
(4) Oyster Creek (POL BWR)

(5) Ginna (POL PWR)
(6) LaCrosse (POL BWR)
(7) Big Rock Point (FTOL BWR)
(8) Palisades (POL PWR)
(9) Dresden 1 (FTOL BWR)
(10) Dresden 2 (POL BWR)
(11) San Onofre (POL PWR)

The SEP review of Dresden Unit 1 has been deferred because the plant is undergoing an extensive modification and is not scheduled for restart before June 1986. Therefore, the total number of plants being reviewed for Phase II is 10.

1.3 Description of Plant

The Dresden Nuclear Generating Station, Unit 2, located in Grundy County, Illinois, is a boiling-water reactor designed by General Electric. The licensee is the Commonwealth Edison Company (CECo). CECo filed the application for a construction permit and operating license in April 1965. The construction permit was issued on January 10, 1966. The initial submittal of the Final Safety Analysis Report was filed on November 17, 1967, and the initial provisional operating license was issued on December 22, 1969. In November 1972 the licensee applied for a full-term operating license. The licensed thermal power rating currently is 2,527 megawatt-thermal (MWt). The Dresden Unit 2 primary coolant system consists of the reactor vessel, recirculation system, main steam system, and isolation condenser. A diagram of the major components of the primary coolant system is shown in Figure 1.1, and the isolation condenser subsystem is shown in Figure 1.2.

The reactor is a single-cycle, forced-circulation, boiling-water reactor producing steam for direct use in the steam turbine. The reactor vessel contains internal components, which include the necessary equipment for separating steam and water flow paths.

The recirculation system provides for forced flow through the reactor core to facilitate heat removal capability. The system consists of 2 external loops with motor-driven centrifugal pumps and 20 jet pumps located in the reactor pressure vessel. Water that is separated from the steam in the reactor vessel and mixes with water provided by the feedwater system is drawn from outside the core, passes through the recirculation pumps, and is discharged back into the reactor below the core area at high velocity through the jet pumps. The action of the jet pumps mixes the high velocity water with water in the reactor vessel, recirculating the water through the core. This serves to increase the heat removal capability of the water. The water then flows upward through the core where boiling produces a steam-water mixture.

The main steam system directs the steam generated in the reactor vessel to the turbine generator for conversion to electrical power. The steam-water mixture travels from the reactor core, through the steam-separating equipment into the four main steam lines. The steam then passes through the main steam lines to the turbine. Included in the main steam system are the relief and safety valves, which provide overpressure protection for the reactor vessel and associated piping systems. The relief valves are also designed to rapidly depressurize the reactor vessel so that the emergency cooling systems will function. The reactor relief valves are located upstream of the first isolation valve and discharge directly to the pressure-suppression pool; the safety valves are located on the steam lines inside the primary containment and discharge to the drywell atmosphere.

The isolation condenser system will provide reactor core cooling if the reactor should become isolated from the main condenser because of closure of the main steam isolation valves. The isolation condenser operates by natural circulation. During operation steam flows from the reactor, condenses in the tubes of the isolation condenser, and flows back to the reactor by gravity.

The containment systems provide a multibarrier pressure-suppression containment composed of a primary containment, the pressure-suppression system, and a secondary containment, the reactor building.

The primary containment system is designed (1) to provide a barrier that will control the release of fission products to the secondary containment and (2) to rapidly reduce the pressure in the containment resulting from a loss-of-coolant accident. The system consists of a drywell, which houses the reactor vessel and recirculation loops; the pressure-suppression pool, which contains the large volume of water used to condense the accident steam release; and the connecting vent systems. The drywell, which is in the shape of a light bulb and is constructed of steel plate, varies in diameter from 37 ft to 66 ft and is approximately 112 ft high. The shell thickness varies from approximately 3/4 to 2-3/4 in. The pressure-suppression chamber is a steel pressure vessel in the shape of a torus with an inside diameter of 30 ft, a water volume of approximately 112,000 cubic feet, and an air volume of approximately 117,000 cubic feet.

The reactor building is designed to provide containment during reactor refueling and maintenance operations when the primary containment system is open. The building will also provide secondary containment when the primary containment is required to be in service. The reactor building consists of the monolithic reinforced concrete floors and walls enclosing the nuclear reactor, primary containment, and reactor auxiliaries, and the building superstructure with sealed panel walls and precast concrete roof.

1.4 Summary of Operating History and Experience

The Dresden Unit 2 plant received a provisional operating license on December 22, 1969, achieved initial criticality on January 7, 1970, and began commercial operation in July 1970. The reactor has a licensed thermal power of 2,527 MWt and a design electric rating of 794 megawatt-electric (MWe).

1.4.1 Summary of Oak Ridge National Laboratory Report

To ensure that the plant's operating history, including plant transients, was appropriately evaluated and factored into the NRC staff evaluation, the staff requested the Oak Ridge National Laboratory (ORNL) to perform a detailed review. A copy of the ORNL report is attached as Appendix F.

Table 1.1 presents the Dresden Unit 2 reactor availability and plant capacity factors. From 1970 through 1981, the reactor availability factor averaged 73.5% and the unit capacity factor averaged 53.8%. The reactor availability factor was above average and the unit capacity factor was average. From 1973 through 1980, the reactor availability and unit capacity factors averaged 80.3% and 61.4%, respectively. The values were lower during 1970 and 1971 because of the introduction of spurious signals into the scram circuitry and maintenance outages to perform repairs on the main transformer and on the main turbine. In 1981, the refueling outage that began at the start of the year was extended so that further repairs on the main turbine could be made.

Of the 206 forced shutdowns and power reductions between 1970 and 1981, 68 were identified as design-basis events (DBEs) of 1 of the following 11 types:

(1) turbine trip (20)

(2) loss of normal feedwater (10)

(3) inadvertent closure of main steam isolation valve (MSIV) (9)

(4) feedwater system malfunctions resulting in increased feedwater flow (8)

(5) loss of condenser vacuum (7)

(6) loss of external electric load (5)

(7) single and multiple recirculation pump trips (3)

(8) inadvertent opening of turbine bypass valves resulting in increased steam flow (2)

(9) inadvertent opening of safety relief valve (2)

(10) startup of an idle recirculation pump (1)

(11) control rod maloperation (1)

Two aspects of the DBEs are relative to frequency of occurrence:

(1) The number of feedwater system malfunctions resulting in increased feedwater flow is somewhat higher than that experienced in other plants, with seven of the eight events occurring between 1970 and 1973.

(2) The total number of DBEs taken individually for each type of event, with the exception of the feedwater malfunctions, is consistent with that experienced in other plants.

Of the 68 DBEs identified through 1981, 40 occurred between 1970 and 1973 at a time when Dresden Unit 2 was experiencing problems (either personnel or inherent) with the introduction of spurious signals in various parts of the reactor protection system. In all instances except one, the engineered safety features functioned properly to bring the reactor to a safe shutdown. That event is discussed in detail in Section 1.4.4.

The trend for the number of reportable events submitted by Dresden Unit 2 has remained relatively constant since 1974 with an average of 63 events occurring from 1974 through 1981. For this 8-year period, a low of 47 events was reported in 1980; the peak years were 1977 and 1981 with 75 and 74 events reported, respectively. For 1970 through 1973, an average of 32 events per year were reported. Inherent equipment failures caused 51% of the reportable events. Human error (including administrative, design, fabrication, installation, maintenance, and operator error) either caused or were directly related to 49% of the reportable events. Of the reports involving human error, the three dominant categories were maintenance errors (11.6%), administrative control errors (11.1%), and design errors (9.4%). There is no apparent trend in the causes of the reportable events.

The major contributor to the significant types of events was various equipment and component failures that caused nine of these events. Pipe cracks that were identified from 1974 through 1978 in assorted systems accounted for six of the significant events. Several cracks were found in the coolant recirculation system and recirculation bypass loop; cracks were also found in the core spray system, feedwater system, and containment isolation systems. Pipe cracking is a generic BWR problem and the Pipe Cracking Study Group formed by NRC has indicated stress-corrosion cracking as the cause. Three events were due to human errors--two operator errors and one fabrication error.

Nine types of recurring events were noted during the two segments of operating history review.

Three of the event types (pipe cracks, MSIV failures, and feedwater regulator valve problems) were identified either by Dresden Unit 2 or NRC staff, and corrective measures were undertaken. Three types (diesel generator failures, control rod and rod drive problems, and the radiaoctive waste management/health physics program problems) were identified by Dresden Unit 2 and NRC staff, and corrective actions were taken or are under consideration. These types, however, continue to recur and/or still are of concern. The remaining three types (operator errors, turbine control problems, and high-pressure coolant injection failures) continued to recur through 1981.

Problems with feedwater regulatory valves were limited to a 3-year period from 1973 to 1976. Problems involved seal leaks, breaking of valve stems, and a blown fuse in a control circuit. Low reactor coolant water level resulted in five of the six events, but there were no instances of total loss of feedwater.

There were 44 instances where the diesel generators failed upon demand; 22 of these were failures to start. From 1975 to 1979, the diesel generators failed almost nine times per year compared with an average of two failures per year for other years. Only one failure to start was reported in 1980 and none in 1981. However, in 1981 there were two diesel generator failures (October 23 and December 1) involving check valve failures on the Unit 2/3 diesel generator engine cooling systems (the Unit 2/3 is a single diesel generator that can be electrically aligned to serve either Unit 2 or 3). A similar failure occurred on the Unit 3 diesel on November 19, 1981. Dresden Unit 2 has experienced three instances of loss of emergency power from the diesel generator failures. During one of the three events, one of the diesels restarted and was declared operable.

Control rod and rod drive problems (numbers indicated in parentheses) were prominent in 1970 (5), 1974 (5), 1977 (7), 1980 (8), and 1981 (6). Slow control rod insert time was experienced during the plant's early life; however, a design modification to the control rod drive inner filter corrected this. Uncoupling of one or two control rods dominated the failures for 1974, 1977, and 1980. The failures in 1981 involved excessive rod insert times.

The problems related to the radioactive waste management/health physics program fall into two categories:

(1) Activity limits were exceeded in various radwaste and drain tanks.

(2) A variety of equipment failures resulted in both gaseous and liquid leaks. These failures and the breakdowns in operations that involved health physics considerations provided the potential for exposures to personnel or resulted in exposures to personnel. The first and more significant radiation overexposure occurred on March 5, 1981, when a portion of a large radiation shielding plug was being removed from inside the reactor vessel during maintenance. The individual involved, a contractor employee, received an exposure of 21 rems while guiding a crane in the removal of the shielding plug. The second overexposure involved a contractor employee who received a cumulative radiation exposure of 3.02 rems for the period January 1 to March 20, 1981.

Operator errors, which include control room personnel, auxiliary operators, and maintenance and testing personnel errors (as compared with the broader category of human errors defined previously), directly caused or complicated some 36 forced reactor shutdowns and 20% of the reportable events.

Problems with the turbine control valve and turbine electrohydraulic control (EHC) system occurred in clusters approximately every 3 to 4 years, in 1972, 1975 to 1976, and 1980. The control-valve problems involved inherent failures and steam leaks. The EHC problems involved low oil pressure and oil leaks that led to forced shutdowns.

Fifty reportable events were filed that involved the high-pressure coolant injection (HPCI) system. Fifteen of these represented failures of the HPCI on demand. The principal causes were failures of motor-operated valves, the turbine stop valve, and the isolation valves. The HPCI failures were evenly distributed throughout the operator experience of Dresden Unit 2, although no failures on demand occurred in 1981. Estimates of the failure rate of the HPCI system for Dresden Unit 2 indicate a failure rate several times that predicted in the Reactor Safety Study (WASH-1400) and a factor of two greater than that observed from historical data.

1.4.2 Operating Experience From January 1, 1982 to August 31, 1982

The Dresden Unit 2 reactor availability and capacity factors were high during the January to August 1982 period. The reactor availability factor was 9,510, and the unit availability factor was 93.8. The unit capacity factor was 82.9% of the design electrical rating and 85.1% of the maximum dependable capacity. All of these factors are well above the Dresden Unit 2 lifetime averages.

During this period only one significant outage occurred. The outage was caused by a failure of an HPCI steam isolation valve to close. The failure was caused by moisture leaking through the valve packing that entered the valve motor operator and shorted out the motor. The valve was repacked and the motor replaced. The safety significance of the event was minimal because all other emergency core cooling systems were operable.

Licensee reponse to numerous NRC requirements such as Three Mile Island Task Action Plan, bulletins, circulars, and information notices has been satisfactory in spite of the heavy work load imposed on the station and corporate staff.

1.4.3 Operating and Regulatory Performance Since January 1, 1980

1.4.3.1 Operating Performance

NRC's Systematic Appraisal of Licensee Performance (SALP I) program from July 1, 1979 through June 30, 1980 and SALP II from July 1, 1980 through December 31, 1981 assessed the licensee's performance in relation to Dresden Unit 2 as well as the entire Dresden station.

Licensee performance including response to events and NRC findings has been acceptable. During this period, several significant events - personnel exposure in excess of regulatory limits, identification of water in HPCI steam lines, and diesel-generator failures - resulted in special inspections and/or investigations.

Throughout this period, several management meetings related to Unit 2 were held. Meetings were held on October 31, 1980 to present the SALP I findings and on June 2, 1982 to present the SALP II findings. Several other meetings were held throughout this period before December 1981 to discuss significant events, NRC concerns, or enforcement findings. The details of these meetings are discussed in the respective inspection or SALP reports.

The health physics appraisal conducted in 1980 identified several areas where improvement could be made as well as some noncompliance items. On the basis of followup inspections, the licensee's actions have significantly improved.

The emergency preparedness appraisal in 1981 and subsequent inspection of the licensee's emergency response plan in coordination with the Federal Emergency Management Agency have recently drawn affirmative evaluations.

In April 1982, a special licensee management appraisal team conducted an intensive inspection in the areas of maintenance, design changes and modifications, corrective action system, training of nonlicensed operators, committee activities, and quality assurance audits. This special inspection provided an excellent overview of station management controls, and even though several items of noncompliance resulted, the overall summary was that the station is adequately managed.

1.4.3.2 Regulatory Performance

The regulatory performance of Dresden Unit 2 during this period has been acceptable. One finding, personnel exposures in excess of regulatory limits, resulted in a Severity Level III citation and a civil penalty. The licensee's followup action was sufficient to warrant no additional enforcement actions.

More recently, the discovery of an open primary containment instrument line was detected and reported. A special inspection was conducted and appropriate enforcement action will be taken.

From December 1981 through August 1982, the licensee's regulatory performance has shown a significant overall improvement as evidenced by inspection findings as well as the licensee's response to events.

1.4.4 Significant Event

The one event where engineered safety features failed to function properly occurred on June 5, 1970 and involved a series of multiple failures complicated by operator error and procedural inadequacies. With the reactor undergoing initial startup tests and operating at 75% power (623 MWe), the incident was initiated by a spurious signal generated in the electrohydraulic control of the turbine-generator set that caused the turbine control valve to open further and the steam bypass valves to the condenser to fully open. Within 1 second the turbine tripped and the reactor scrammed. The two operating feedwater pumps tripped because of low suction pressure resulting from the increased feedwater flow. Subsequently, the MSIVs closed and the control of the water level in the pressure vessel became difficult. Water level began rising again, but because the level-indicator chart pen being observed by the operator became stuck, the operator, not knowing the level was still increasing further, increased the flow rate of feedwater. By the time the operator discovered the stuck pen, the water level had risen enough to flood the main steam lines and the isolation condenser steam line. The incident was further complicated at this point by a lack of procedural guidance under conditions of high reactor coolant in the pressure vessel. The continued input of water coupled with afterheat from the reactor core and closure of the main steam line valves caused the pressure-vessel pressure to begin increasing rapidly. The isolation condenser was manually actuated, but was subsequently isolated by the circuitry that is used to detect a break in the condenser return line. The high flow setpoint for this circuitry, specified by Technical Specifications, had been set too low to allow for the high condensate flow that is encountered following startup with the condensate leg filled with cold water. An attempt to reopen the main steam line valves to dump steam through the turbine-bypass valves failed because the valves had not been reset from the earlier trip that had closed them. Following the automatic tripping of the recirculation pumps and automatic startup of the standby diesel generators, the core spray and low-pressure coolant injection systems started but did not inject water because the reactor pressure exceeded the pump head of both systems. The high-pressure coolant injection system started but did not inject water because it had been valved out earlier for repairs after its backup systems had been proof-tested as provided for in the Technical Specifications. Actual water injection by this system would have been automatically inhibited by the high-water signal from the pressure vessel water-level monitors. With the isolation condenser inoperable, the operator manually opened a pressure-relief valve several times throughout the incident so that steam could be dumped to the pressure-suppression pool to reduce the pressure in the pressure vessel and thereby remove the reactor decay heat. Two-phased flow caused a waterhammer that resulted in opening a relief valve to the drywell. The discharge from the relief valve impinged on two other relief valves causing them to open. They remained open until they were closed manually after the vessel was depressurized and cooled down. Several thousand gallons of primary water leaked to the drywell. The containment zone was contaminated, but no measurably radioactivity was released to the site or the environs. Damage to the plant was minor.

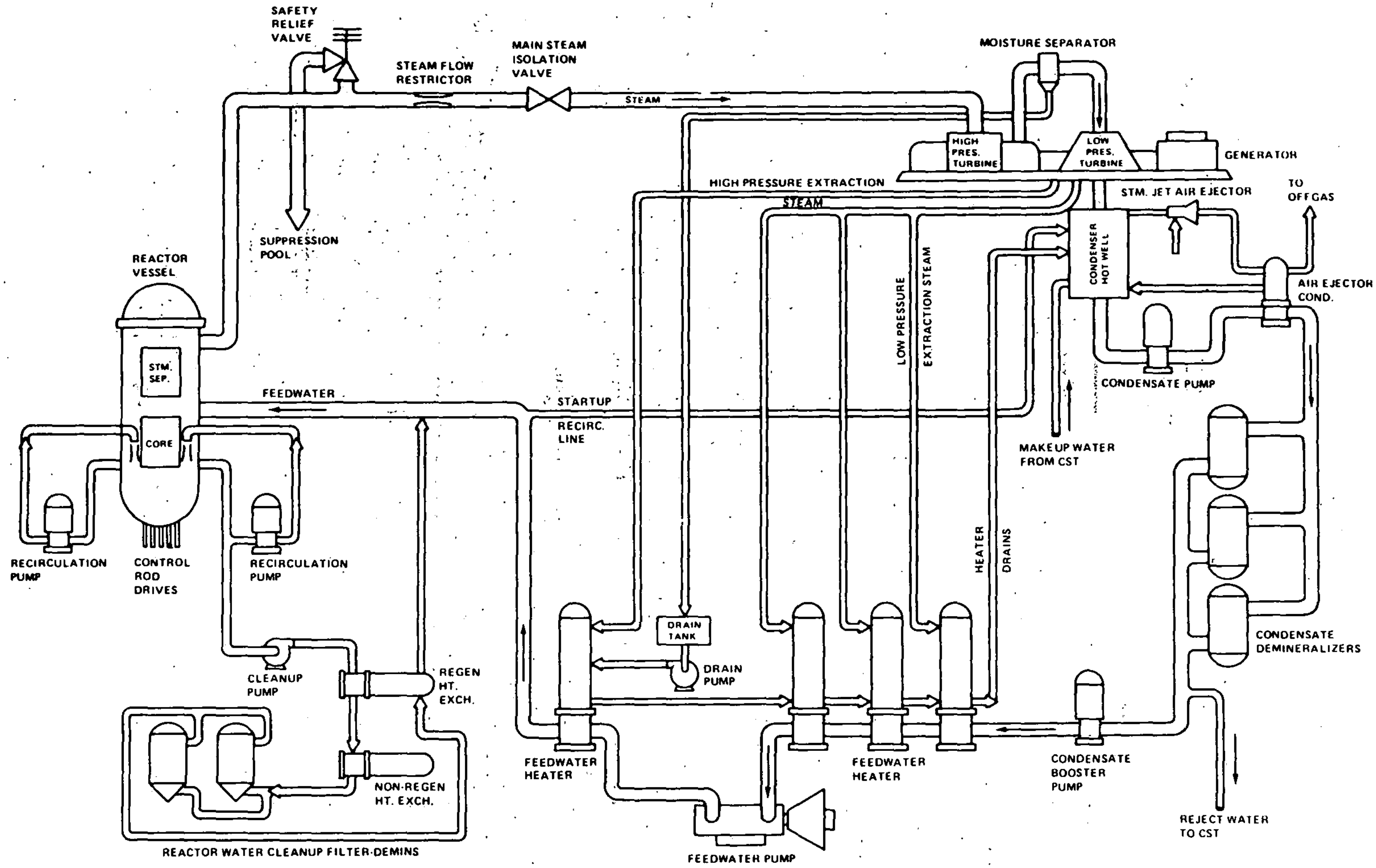

Figure 1.1 Dresden Unit 2 - primary coolant system

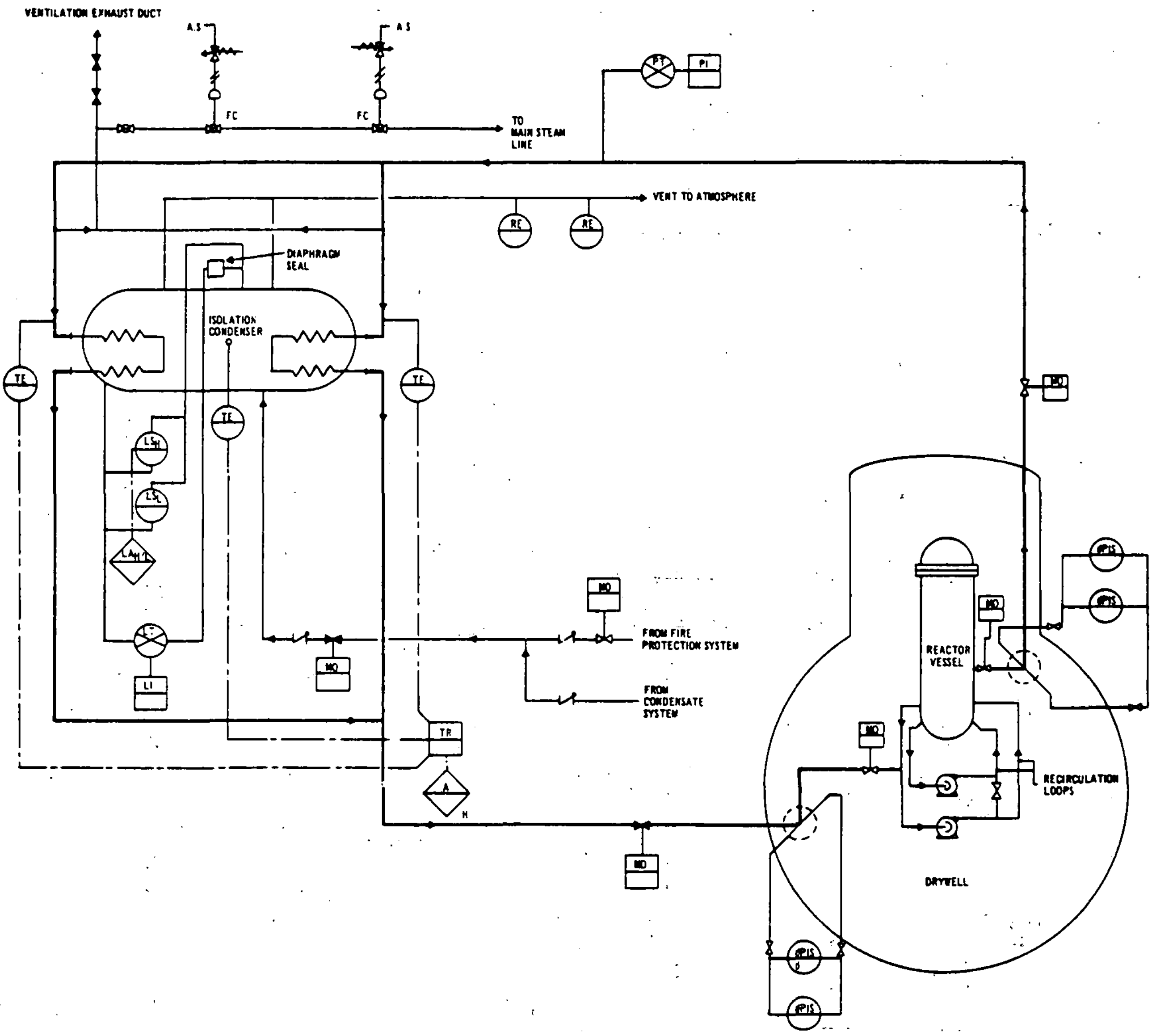

Figure 1.2 Isolation condenser subsystem
Source: Dresden FSAR Figure 4.5.1

Table 1.1 Dresden Unit 2 availability and capacity factors

Factor	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	Avg
Reactor availability	44.5	69.3	62.5	90.8	66.8	57.8	78.3	73.5	96.3	83.2	95.6	62.8	73.5
Unit availability	34.0	65.0	59.8	87.6	64.1	55.1	75.9	71.9	94.2	81.6	93.3	60.1	70.2
Unit capacity (MDC)	ND	ND	ND	ND	ND	42.3	64.5	52.2	8[illegible]	73.0	67.6	50.4	62.1
Unit capacity (DER)	22.4	37.3	45.2	70.8	48.2	41.2	61.6	50.8	82.0	71.0	65.7	49.0	53.8

Notes: MDC = maximum dependable capacity (772 MWe); DER = design electrical rating (794 MWe); ND = No data.

2 REVIEW METHOD

2.1 Overview

The Systematic Evaluation Program (SEP) review procedure represents a departure from the typical NRC staff reviews conducted to support the granting of a construction permit or operating license for a new facility or a license amendment for an operating facility. A typical licensing review starts with the submittal by the utility of a safety analysis report (SAR) that describes the design of the proposed plant. The staff reviews the SAR on the basis of the Standard Review Plan (SRP), Regulatory Guides, and Branch Technical Positions (found in SRP) that constitute current licensing criteria. The guidelines in the SRP represent acceptable means of complying with licensing regulations specified in Title 10 of the Code of Federal Regulations (10 CFR).

The SEP was initiated by NRC, and not by the licensee as part of an application for a license or request for a license amendment. The SEP procedure involves several phases of data gathering and evaluation so that an integrated assessment of the overall plant safety can be made. The various phases and their interrelationships are described below.

2.2 Selection of Topic List

A list of significant safety topics was derived from existing safety issues during Phase I of the program. More than 800 items were considered in the development of the original list; however, a number of these were found to be duplicative in nature or were deleted for other reasons. Categories of topics that were deleted for other reasons are (1) those not normally included in the review of light-water reactors, (2) those related either to research-and-development programs or to the development of analytical evaluation models and methodology, and (3) those that are reviewed on a periodic basis in accordance with current criteria (for example, fuel performance). The topics retained numbered 137; these were arranged in groups corresponding to the organization of the SRP. A "definition" was prepared for each topic to ensure a common understanding. This definition plus a statement of the safety objective for the review and the status of the review at that time is contained in Appendix A for ease of reference.

During the course of this review, the number of topics that applied to all plants was reduced further because some topics were being reviewed generically under either the Unresolved Safety Issues (USIs) program or the Three Mile Island (TMI) NRC Action Plan; also, duplicates found within the SEP topics were deleted. Appendix B shows these topics along with the corresponding USI, TMI task, or SEP topic referenced. The basis for deletion appears in Appendix A under the individual topics. The current status of USI and TMI Action Plan Item reviews applicable to SEP will be discussed in a POL conversion safety evaluation report that will be issued following completion of the integrated assessment.

Plant-specific deletions other than those common to all SEP plants were made to account for nonapplicability of particular topics to Dresden Unit 2. The plant-specific topics that were removed for Dresden Unit 2 and the basis for deletion are shown in Appendix C.

For Dresden Unit 2, this process resulted in 88 topics from the topic list that formed the SEP review. The final list of 88 topics that were reviewed appears in Section 3.1.

The milestones in the review of the SEP program and the Dresden Unit 2 plant are shown in Table 2.1.

2.3 Topic Evaluation Procedures

Each SEP topic in Section 3.1 was reviewed to determine whether the corresponding plant design was consistent with current licensing criteria such as regulations, guides, and SRP review criteria, or the equivalent of such criteria. Safety evaluation reports (SERs) for all 88 topics were issued to document the comparison with current licensing criteria and to identify potential areas for backfitting. References for letters regarding the individual topic SERs are contained in Appendix E. These documents describe the detailed evaluations where conclusions are summarized in this report.

Topics were evaluated by one of two methods:

(1) The NRC staff reviewed and formally issued an SER to the licensee. This SER was termed a draft because it was only one input element to the evaluation. The purpose of the draft SER was to verify the factual accuracy of the described facility and to allow the licensee to identify possible alternate approaches to meeting the current licensing criteria. After a review of the licensee's comments on the draft SER, factual changes were incorporated as needed, proposed alternatives were reviewed, and the SER was issued in final form.

(2) The licensee submitted a safety analysis report and the staff issued a final SER based on a review of this submittal.

After completion of the topic evaluation, the disposition of each topic was grouped according to one of the following results:

(1) The plant is consistent with current licensing criteria and the topic review is considered complete. If the plant does not meet current licensing criteria, but the present design is equivalent to current criteria, the topic is also considered complete. A justification for this conclusion is provided in the topic SER. The topics in this category are identified in Section 3.1 of this report by an asterisk.

(2) The plant is not consistent with current licensing criteria, but the licensee has implemented design or procedural changes that the staff finds acceptable. Although the licensee committed to certain design or procedural changes during the course of the topic reviews for Dresden Unit 2, none were implemented so that differences were resolved during a topic review. Consequently, none of the topics fell into this category.

(3) The plant is not consistent with current licensing criteria, and the differences from these criteria are to be evaluated as potential candidates for backfitting. If the staff determines the difference is of immediate safety significance, action is taken to resolve the issue promptly. No issues at Dresden Unit 2 required that prompt action be taken. If the difference is not of immediate safety significance, the resolution is deferred to the integrated plant safety assessment to obtain maximum benefit from coordinated and integrated backfitting decisions. The SEP evaluation of all 88 topics led to the conclusion that 34 topics were not consistent with current licensing criteria. All of these topics were considered in the integrated safety assessment and appear in Section 4.

2.4 Integrated Plant Safety Assessment

The objective of the integrated plant safety assessment is to make balanced and integrated decisions on backfitting current licensing criteria to SEP facilities. Factors considered important in reaching decisions on backfitting include safety significance, radiation exposure to workers, and, to a lesser extent, implementation impact and schedule.

A meeting was held with the licensee to discuss these factors as they related to the differences identified during the SEP review between actual facility design and current licensing criteria and to obtain the licensee's views on safety significance and possible corrective actions.

These factors were considered in reaching a decision on backfitting and are discussed in Section 4 for each identified difference between actual facility design and current licensing criteria. Because these factors sometimes rely on judgment, risk assessment techniques were used to the extent possible to supplement the staff's judgments concerning safety significance. The probabilistic risk assessment (PRA) performed by Sandia National Laboratories, along with comments by the staff, appears in Appendix D. For reasons given in Appendix D, only certain topics could be readily analyzed by a PRA. Of a total number of 34 topics considered in the integrated assessment, 20 were evaluated using PRA techniques.

Table 2.1 Topic list selection and resolution

ORIGINAL PHASE I TOPIC LIST
800

↓

Many of these topics were deleted because they were duplicative in nature, were not normally included in the review of light-water reactors, were related to research-and-development programs, or were reviewed on a periodic basis in accordance with current criteria.

↓

FINAL LIST OF PHASE I TOPICS REVIEWED DURING PHASE II
137 (see Appendix A)

↓

Of the 137 topics, 19 were deleted because they were being reviewed generically under either the Unresolved Safety Issues (USIs) program or the Three Mile Island (TMI) NRC Action Plan (see Appendix B).

↓

REMAINING TOPICS AFTER DELETION OF USIs AND TMI-RELATED TOPICS
118

↓

Of the remaining 118 topics, 30 were deleted because the topics did not apply to Dresden Unit 2 (see Appendix C).

↓

FINAL NUMBER OF TOPICS REVIEWED FOR DRESDEN UNIT 2
88 (see Section 3.1 and Appendix E)

↓

TOPICS THAT MET CURRENT CRITERIA OR WERE ACCEPTABLE ON ANOTHER DEFINED BASIS
54 (see Section 3.1)

↓

TOPICS THAT MET CURRENT CRITERIA OR WERE ACCEPTABLE ON ANOTHER DEFINED BASIS AFTER MODIFICATIONS MADE DURING TOPIC REVIEW
0

↓

TOPICS CONSIDERED FOR BACKFIT IN THE INTEGRATED ASSESSMENT
34 (see Table 4.1 and Sections 4.1-4.32)

3 TOPIC EVALUATION SUMMARY

3.1 Final Dresden Unit 2-Specific List of Topics Reviewed

Listed below are the 88 topics that were reviewed for Dresden Unit 2. The topics with asterisks are those for which the plant meets current criteria or was acceptable on another defined basis:

TOPIC	TITLE
II-1.A*	Exclusion Area Authority and Control
II-1.B*	Population Distribution
II-1.C*	Potential Hazards or Changes in Potential Hazards Due to Transportation, Institutional, Industrial, and Military Facilities
II-2.A*	Severe Weather Phenomena
II-2.C*	Atmospheric Transport and Diffusion Characteristics for Accident Analysis
II-3.A*	Hydrologic Description
II-3.B	Flooding Potential and Protection Requirements
II-3.B.1	Capability of Operating Plant To Cope With Design-Basis Flooding Conditions
II-3.C	Safety-Related Water Supply (Ultimate Heat Sink (UHS))
II-4*	Geology and Seismology
II-4.A*	Tectonic Province
II-4.B*	Proximity of Capable Tectonic Structures in Plant Vicinity
II-4.C*	Historical Seismicity Within 200 Miles of Plant
II-4.D*	Stability of Slopes
II-4.E*	Dam Integrity
II-4.F*	Settlement of Foundations and Buried Equipment
III-1	Classification of Structures, Components, and Systems (Seismic and Quality)
III-2	Wind and Tornado Loadings

TOPIC	TITLE
III-3.A*	Effects of High Water Level on Structures
III-3.C	Inservice Inspection of Water Control Structures
III-4.A	Tornado Missiles
III-4.B	Turbine Missiles
III-4.C*	Internally Generated Missiles
III-4.D*	Site-Proximity Missiles (Including Aircraft)
III-5.A	Effects of Pipe Break on Structures, Systems, and Components Inside Containment
III-5.B	Pipe Break Outside Containment
III-6	Seismic Design Considerations
III-7.B	Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria
III-7.D*	Containment Structural Integrity Tests
III-8.A	Loose-Parts Monitoring and Core Barrel Vibration Monitoring
III-8.C*	Irradiation Damage, Use of Sensitized Stainless Steel, and Fatigue Resistance
III-10.A	Thermal-Overload Protection for Motors of Motor-Operated Valves
III-10.C*	Surveillance Requirements on BWR Recirculation Pumps and Discharge Valves
IV-1.A*	Operation With Less Than All Loops in Service
IV-2*	Reactivity Control Systems Including Functional Design and Protection Against Single Failures
IV-3*	BWR Jet Pump Operating Indications
V-4*	Piping and Safe-End Integrity
V-5	Reactor Coolant Pressure Boundary (RCPB) Leakage Detection
V-6	Reactor Vessel Integrity
V-10.A*	Residual Heat Removal System Heat Exchanger Tube Failures
V-10.B	Residual Heat Removal System Reliability

TOPIC	TITLE
V-11.A	Requirements for Isolation of High- and Low-Pressure Systems
V-11.B	Residual Heat Removal System Interlock Requirements
V-12.A*	Water Purity of BWR Primary Coolant
VI-1*	Organic Materials and Postaccident Chemistry
VI-2.D*	Mass and Energy Release for Postulated Pipe Break Inside Containment
VI-3*	Containment Pressure and Heat Removal Capability
VI-4	Containment Isolation System
VI-6	Containment Leak Testing
VI-7.A.3*	Emergency Core Cooling System Actuation System
VI-7.A.4	Core Spray Nozzle Effectiveness
VI-7.C*	Emergency Core Cooling System (ECCS) Single-Failure Criterion and Requirements for Locking Out Power to Valves, Including Independence of Interlocks on ECCS Valves
VI-7.C.1	Appendix K--Electrical Instrumentation and Control Re-Reviews
VI-7.C.2*	Failure Mode Analysis (Emergency Core Cooling System)
VI-7.D*	Long-Term Cooling Passive Failures (e.g., Flooding of Redundant Components)
VI-10.A	Testing of Reactor Trip System and Engineered Safety Features, Including Response-Time Testing
VI-10.B	Shared Engineered Safety Features, Onsite Emergency Power, and Service System For Multiple-Unit Stations
VII-1.A	Isolation of Reactor Protection System From Nonsafety Systems, Including Qualification of Isolation Devices
VII-1.B*	Trip Uncertainty and Setpoint Analysis Review of Operating Data Base
VII-2*	Engineered Safety Features System Control Logic and Design
VII-3	Systems Required for Safe Shutdown
VII-6*	Frequency Decay
VIII-1.A*	Potential Equipment Failures Associated With Degraded Grid Voltage

TOPIC	TITLE
VIII-2	Onsite Emergency Power System (Diesel Generator)
VIII-3.A	Station Battery Capacity Test Requirements
VIII-3.B	DC Power System Bus Voltage Monitoring and Annunciation
VIII-4*	Electrical Penetrations of Reactor Containment
IX-1*	Fuel Storage
IX-3*	Station Service and Cooling Water Systems
IX-5	Ventilation Systems
IX-6*	Fire Protection
XIII-2*	Safeguards/Industrial Security
XV-1	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve
XV-3*	Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulator Failure (Closed)
XV-4*	Loss of Nonemergency AC Power to the Station Auxiliaries
XV-5*	Loss of Normal Feedwater Flow
XV-7*	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break
XV-8*	Control Rod Misoperation (System Malfunction or Operator Error)
XV-9*	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate
XV-11*	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (BWR)
XV-13*	Spectrum of Rod Drop Accidents (BWR)
XV-14*	Inadvertent Operation of Emergency Core Cooling System and Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory
XV-15*	Inadvertent Opening of a PWR Pressurizer Safety/Relief Valve or a BWR Safety/Relief Valve

TOPIC	TITLE
XV-16	Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment
XV-18	Radiological Consequences of Main Steam Line Failure Outside Containment
XV-19*	Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary
XV-20*	Radiological Consequences of Fuel-Damaging Accidents (Inside and Outside Containment)
XVII*	Operational Quality Assurance Program[1]

3.2 Topics for Which Plant Design Meets Current Criteria or Was Acceptable on Another Defined Basis

As listed in Section 3.1.

3.3 Topics for Which Plant Design Meets Current Criteria or Equivalent Based on Modifications Implemented by the Licensee

During the topic reviews for Dresden Unit 2, the licensee committed to certain design changes, procedural changes, or analyses to resolve differences identified. However, none of these actions were implemented so that no topic was considered resolved before the integrated assessment. Consequently, all of the differences identified during the topic reviews and the commitments made by the licensee are discussed in the context of the integrated assessment in Section 4.

[1]The Operational Quality Assurance Program was reviewed according to the criteria specified for operating reactors in 1974 (see Appendix A). NRC is currently evaluating all aspects of Nuclear Power Plant Quality Assurance Programs. Additional review of this issue will be performed outside the context of SEP.

4 INTEGRATED ASSESSMENT SUMMARY

Table 4.1 contains the list of topics considered in the integrated assessment, whether Technical Specification requirements or backfit are needed, and whether or not the licensee proposes to backfit. A more detailed description of each topic with identified differences follows.

4.1 Topic II-3.B, Flooding Potential and Protection Requirements; Topic II-3.B.1, Capability of Operating Plants To Cope With Design-Basis Flooding Conditions; Topic II-3.C, Safety-Related Water Supply (Ultimate Heat Sink (UHS))

10 CFR 50 (GDC 2), as implemented by SRP Sections 2.4.2, 2.4.5, 2.4.10, 2.4.11 and 3.4.1 and Regulatory Guides 1.27 and 1.59, requires that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as flooding. The safety objective of these topics (II-3.B, II-3.B.1, and II-3.C) is to verify adequate operating procedures and/or system design are provided to cope with the design-basis flood.

The site grade elevation is 517 ft mean sea level (MSL). During the staff's review of the hydrology-related topics, the following flooding elevations were identified, as defined by current licensing criteria:

probable maximum flood (PMF)
still water - 525 ft MSL
wave runup - 528 ft MSL

As a result of these flooding levels, the staff has identified the following issues.

4.1.1 Design-Basis Groundwater Level

The original design value for groundwater level at Dresden Unit 2 was 514 ft MSL. Actual plant grade is at 517 ft MSL. However, a 3-ft change in ground water level is not a significant change in structural loading when combined with other loads such as seismic. Also, as part of its review of Topic III-3.A, "Effects of High Water Level on Structures," the staff concluded that structural integrity would be maintained for water levels up to 517 ft MSL. Therefore, based on margins in structures under postulated seismic loadings (Topic III-6, "Seismic Design Considerations") and adequate margin for static loading, the staff concludes that further analysis of the effects of groundwater level on structures is not warranted. Backfitting is not required.

4.1.2 Probable Maximum Flood

The staff has calculated the probable maximum flood for the Dresden site to be 528 ft MSL, including wave runup. The Dresden Unit 2 plant is not protected to the PMF level as required by current licensing criteria.

In addition, the staff has determined that the expected 100-year water surface elevation and standard project flood will flood the service water pump motors. It is the staff's position that the licensee ensure the capability to achieve and maintain a safe shutdown condition for all expected flooding levels. The protection features should be addressed in plant emergency procedures. These procedures are discussed in Section 4.1.4.

4.1.3 Roof Loadings

The roofs of safety-related structures (turbine and reactor buildings and crib house) have not been designed to sustain the loading associated with the probable maximum precipitation (PMP). The design of the roof parapets will allow substantial accumulation of ponded water. The staff's position is that inservice inspection of roof drains is not a feasible solution because of the potential for blockage by debris and the frequency of inspection necessary to ensure drain capacity.

By letter dated November 17, 1982(a), the licensee has committed to make structural modifications to the roof parapets (scuppers) to ensure that loadings resulting from ponded water will be within the structural capability of the roof.

4.1.4 Flood Emergency Plan

The licensee's flood emergency plan in its present form does not meet current criteria regarding its adequacy to provide for safe shutdown of the facility following a severe river flood. Among the items identified during the topic review, the staff considers the following concerns to be significant:

(1) The procedures relied on the capability of the licensee to predict floods sufficiently in advance to provide the time necessary to get the plant to cold shutdown. The licensee does not have the professional staff with the hydrologic experience necessary to devise and implement a river forecast system and elaborate flood emergency plan.

(2) On the basis of the computed hydrograph information (i.e., flood stage versus time for a PMF), there is not sufficient time to get the plant to cold shutdown using normal shutdown procedures. The emergency plan does not address other procedures that would be required in a limited time frame.

(3) The emergency plan does not adequately address postflood conditions such as sources of emergency cooling water, time required to return safety systems to service, and fuel requirements and availability for diesel- and gasoline-powered equipment.

The existing flood emergency plan does not provide assurance that the ultimate heat sink can provide for a safe plant shutdown. Scenarios exist that would result in the flooding of safety equipment. Also, failure of the Dresden Island Lock and Dam would result in low water levels that may affect the capability of the ultimate heat sink. Further, plant procedures require internal flooding of structures, which could result in a loss of all reactor cooling.

The staff has recommended that the licensee have the capability to install and operate an emergency pump above the PMF level capable of providing 100% makeup water to the isolation condenser and other cooling needs for the duration of the flood, including the time needed to restore the operation of flood-damaged components. The plant currently has the capability to use a portable pump to supply cooling water directly to the isolation condenser using a fire hose connection.

By letter dated November 17, 1982(a), the licensee committed to revise the flood emergency procedure. Included in these changes will be flood predictions, gasoline pump connections and fuel supplies, intake canal level gauge, and clearer direction on the use of instruments.

On December 3-4, 1982, the Dresden site was subjected to flood waters that exceeded 509 ft MSL and resulted in the shutdown of both units. NRC staff representatives were at the site for most of the event. The NRC observers have recommended action to be taken by the licensee. The Office of Inspection and Enforcement will ensure that the revised flood emergency plan is sufficient to cope with a design-basis-flood event.

4.2 Topic III-1, Classification of Structures, Components, and Systems (Seismic and Quality)

10 CFR 50 (GDC 1), as implemented by Regulatory Guide 1.26, requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of safety functions to be performed. The codes used for the design, fabrication, erection, and testing of the Dresden Unit 2 plant were compared with current codes.

The development of the current edition of the American Society of Mechanical Engineers "Boiler and Pressure Vessel Code" (ASME Code) has been a process evolving from earlier ASME Code, American National Standards Institute, and other standards, and manufacturer's requirements. In general, the materials of construction used in earlier designs provide comparable levels of safety.

The review of this topic identified several systems and components for which the licensee was unable to provide information to justify a conclusion that the quality standards imposed during plant construction meet quality standards required for new facilities. The staff did not identify any inadequate components. However, because of the limited information on the components involved, the staff was unable to conclude that for code and standard changes deemed important to safety, the Dresden Unit 2 plant met current requirements.

It is the staff's position that the licensee provide the radiography requirements and demonstrate adequate fracture toughness for the components identified or demonstrate that the consequences of their failure are acceptable.

4.2.1 Radiography Requirements

ASME Code, Section III, requires that Category A, B, and C weld joints be radiographed. Furthermore, ASME Code, Section III, 1977 Edition, requires that weld joints for Class 1 and 2 piping, pumps, and valves be radiographed.

The staff has reviewed, on a sample basis, the fabrication and construction inspection program implemented at Dresden Unit 2. It finds that the program is generally in agreement with current requirements except that the following items have not been addressed:

(1) Class 2 vessels built to Class C requirements and containing Category C joints, along with the examination technique employed, should be identified.

(2) The actual examination given to the recirculation system pump casing (this is a Class 1 component built to Class C requirements) should be described.

4.2.2 Fracture Toughness

ASME Code, Section III, requires fracture toughness testing of pressure-retaining material and material welded thereto. The staff's safety evaluation forwarded by letter dated September 2, 1982(a) has identified the following areas where the fracture toughness requirements have not been provided:

(1) <u>Reactor Shutdown Cooling System (RSCS), Reactor Building Closed Cooling Water (RBCCW) System, and Reactor Water Cleanup (RWCU) System</u>

The licensee has indicated that fracture toughness testing data do not exist for the RSCS, RBCCW, and RWCU systems. The RSCS is designed to cool the reactor water when the temperature and pressure in the reactor fall below the point at which the main condenser can no longer be used. If any of the system design limits are exceeded, automatic interlocks will prevent the system from being put into operation. If the RSCS were inoperable or unavailable for any reason, the safety-grade low-pressure coolant injection (LPCI) and core spray systems could be used to inject cooling water into the reactor core. Therefore, the staff has concluded that the RSCS is not required to bring the plant to a safe shutdown condition.

The RBCCW system is designed to provide cooling for equipment and systems in the reactor building. The system is not required to perform any postaccident heat removal functions. The staff's evaluation of SEP Topic IX-3, provided by letter dated June 30, 1981, has determined that RBCCW flow to the equipment cooled by the RBCCW can be lost under both normal and postaccident conditions, and although operator action may be required to restore flow to continue plant operation, the consequences are of little safety concern. Therefore, the staff has concluded that the RBCCW system is not important to safety.

The RWCU system is designed to remove impurities from the reactor coolant system and is not required for any safe shutdown or postaccident function. The RWCU system does form part of the reactor coolant pressure boundary. Failures of this system are discussed in Section 4.16 regarding Topic V-II.A, "Requirements for Isolation of High- and Low-Pressure Systems."

Because of the low safety significance of the system and the cost of obtaining and testing samples of the systems, the staff has concluded that fracture toughness testing is not necessary and backfitting is not required.

(2) <u>System Components</u>

For many components identified as requiring fracture toughness testing, the licensee has not provided the actual requirements imposed or the test results requested in the staff's draft evaluation forwarded by letter dated March 9,

1982. This information is necessary to complete the staff's evaluation because of the radical change in fracture toughness test requirements that occurred in 1972 and the importance of adequate fracture toughness to ensure the integrity of the reactor coolant pressure boundary and safe shutdown and accident-mitigating systems. The staff is lacking the necessary information for the following components:

(a) core spray system - pump casing

(b) low-pressure coolant injection/containment cooling - pump casing, shell side of heat exchanger

(c) high-pressure coolant injection - pump casing; piping, fittings, and valves

(d) condensate/feedwater system - piping from reactor vessel to outermost containment isolation valve

(e) main steam system - piping, fittings, and valves

4.3 Topic III-2, Wind and Tornado Loadings

10 CFR 50 (GDC 2), as implemented by SRP Sections 3.3.1 and 3.3.2 and Regulatory Guides 1.76 and 1.117, requires that the plant be designed to withstand the effects of natural phenomena such as wind and tornadoes.

The existing design and construction of structures important to safety do not meet current licensing criteria regarding the ability of safety-related structures to resist tornado winds of 360 mph and differential pressures of 3.0 psi. The following were identified by the staff as items not meeting the prescribed loads.

4.3.1 Reactor Building Structure Above the Operating Floor

The windspeed capacity for the reactor building steel structure and siding above the operating floor is lower than those required by the site-specific tornado-imposed loads. In particular, the staff consultants have calculated that the limiting structural elements of the east-side steel columns and siding have a load capacity of 160 and 170 to 190 mph, respectively, and the south-side columns are able to withstand a 280-mph windspeed.

The only safety-related system located on the main floor of the reactor building is the spent fuel pool. The safety concern would relate to failure of the steel columns in such a way that they would enter the spent fuel pool and damage the spent fuel assemblies. Even in the unlikely event that fuel is sufficiently damaged to allow the release of radioactive gas to the pool, any subsequent release from the pool would be rapidly dispersed because of the turbulent nature of tornado winds. Therefore, the radiological consequences would be inconsequential. Further, the staff's evaluation of SEP Topic II-2.A, "Severe Weather Phenomena," provided a probability analysis of expected tornado windspeeds. The study showed the mean probability of exceeding a windspeed of 160 mph to be approximately 3×10^{-5} per year and the mean probability of exceeding 280 mph to be approximately 5×10^{-7} per year.

It is the staff's judgment that the combined probabilities of exceeding a tornado wind speed of 160 MPH with the probability of the failed structure entering the spent fuel pool in such a way as to cause fuel damage are sufficiently low together with the limited radiological consequences so that there is no need to upgrade the reactor building structure. Backfitting, therefore, is not required.

4.3.2 Ventilation Stack

The stack capacities provided to the staff by the licensee are lower (255 mph) than those required by the site-specific tornado-imposed loads. Failure of the stack could affect the integrity of seismic Category I structures because the stack is in close proximity to these structures.

By letter dated November 22, 1982, the licensee provided the staff with information regarding the analysis methods used to calculate the stack wind capacity. The licensee stated that the analysis was performed using the American Concrete Institute (ACI) ultimate strenth method (ACI Std. 307-1979). The staff currently accepts only working stress analysis to determine chimney capacities. Futher, ACI Std. 307-1979, "Specification for the Design and Construction of Reinforced Concrete Chimneys," excludes ultimate strength design because of the lack of experimental data on hollow concrete cylinders.

Therefore, it is the staff position that the licensee provide further information regarding the stack wind capacity and/or demonstrate that stack failure will not prevent achieving and maintaining safe shutdown.

4.3.3 Components Not Enclosed in Qualified Structures

The staff's analysis did not include the systems and components important to safety that are located outside qualified structures. It is the staff's position that the licensee identify those components and ensure that they are designed to withstand the postulated tornado loading or that their loss of function will not adversely affect safe operation of the plant.

By letter dated November 22, 1982, the licensee supplied information regarding those components not enclosed within qualified structures. The staff is currently reviewing the licensee's response.

4.3.4 Roof Decks

Roof decks consisting of builtup roofing as opposed to structural roof slabs made of concrete were not investigated by the staff. It is expected that such roof decks will have minimal resistance to differential pressure.

By letter dated November 22, 1982, the licensee supplied information regarding the structural capability of the roof decks. The staff is currently reviewing this information.

4.3.5 Load Combinations

As a result of the topic review, the staff was unable to determine if straight wind loads (not tornado loads) were combined with other loads (i.e., operating pipe reaction loads and thermal loads). The effect of combining wind loads with other loads is addressed with SEP Topic III-7.B, "Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria."

4.4 Topic III-3.C, Inservice Inspection of Water Control Structures

10 CFR 50 (GDC 2, 44, and 45), as implemented by Regulatory Guide 1.127, requires that structures, systems, and components important to safety be designed to withstand natural phenomena such as floods and that a system to transfer heat to an ultimate heat sink be provided. The inspection is intended for water control structures used for flood protection (on or off site) and emergency cooling water systems. The safety objective is to ensure that water control structures that are part of the ultimate heat sink are available at all times during both normal and accident conditions. The topic review identified several items for which current acceptance criteria are not met.

4.4.1 Flow-Regulation Station

The staff found that the period of inspection for the flow-regulation station electrical and mechanical equipment does not comply with current licensing criteria.

By letter dated September 2, 1982(b), the licensee has stated that the flow-regulation station is not safety related and that a specific inspection frequency is not necessary. Failure of the station would be in the as-is configuration and plant operation could continue in the failed mode.

On the basis of the above, the staff has concluded that the flow-regulation station is not safety related. Therefore, backfitting is not required.

4.4.2 Intake and Discharge Structures

The staff found that the inspection frequency for the structural integrity of the intake and discharge structures does not meet current criteria, since this frequency is based on observed sedimentation rates rather than historical occurrences of vertical and horizontal movement that could affect the stability of the structures. Agreement has been reached regarding inspection frequency as discussed in Section 4.4.3.

The staff's evaluation of Topic II-4.D, "Stability of Slopes," has concluded that the rock into which the canals are cut is sound and capable of maintaining a stable vertical cut slope under normal and earthquake conditions. In addition, since the plant structures are founded on sound rock, as discussed under SEP Topic II-4.F, settlement of plant structures is not a problem. Therefore, backfitting is not required.

4.4.3 Inspection Program

The inspection program does not comply with current criteria because the program is not overseen by qualified engineering personnel who would document the results of inspections. In addition, the inspection program should be formalized so that provisions will exist for special inspections immediately after the occurrence of extreme events.

By letter dated September 2, 1982(b), the licensee stated that procedural changes to ensure review and approval of the inspection program by qualified engineering personnel and to initiate inspection after extreme events (including floods, severe icing, earthquakes, etc., and unusual structural behavior such as concrete deterioration or cracking) have been initiated by the Dresden station. The licensee has committed to modify the existing plant procedures in accordance with staff recommendations.

4.5 Topic III-4.A, Tornado Missiles

10 CFR 50 (GDC 2), as implemented by Regulatory Guide 1.117, prescribes structures, systems, and components that should be designed to withstand the effects of a tornado, including tornado missiles, without loss of capability to perform their safety functions. Regulatory Guide 1.117 requires that structures, systems, and components that should be protected from the effects of a design-basis tornado are (1) those necessary to ensure the integrity of the reactor coolant pressure boundary, (2) those necessary to ensure the capability to shut down the reactor and maintain it in a safe shutdown condition (including both hot standby and cold shutdown), and (3) those whose failure could lead to radioactive releases resulting in calculated offsite exposures greater than 25% of the guideline exposures of 10 CFR 100 using appropriately conservative analytical methods and assumptions. The physical separation of redundant or alternate structures or components required for the safe shutdown of the plant is not considered acceptable by itself for providing protection against the effects of tornadoes, including tornado-generated missiles, because of the large number and random direction of potential missiles that could result from a tornado as well as the need to consider the single-failure criterion. The staff has found that there are portions of safety-related systems that are not protected from tornado missiles.

4.5.1 Service Water System (SWS)

The staff has determined that the service water supply for two ventilation systems necessary for safe shutdown is not protected from tornado missiles. These systems are (1) the control room ventilation system and (2) the auxiliary electrical equipment room ventilation system.

(1) Control Room Ventilation System

Portions of the SWS necessary for safe operation of the control room ventilation system are located in a non-tornado-missile-protected section of the crib house.

NUREG-0737, "Clarification of TMI Action Plan Requirements," Item III.D.3.4, "Control Room Habitability Requirements," states that all licensees should provide assurance that the habitability systems will operate under all postulated conditions. Implementation of the TMI Action Plan is being conducted independently of the SEP. Therefore, backfitting is not required.

(2) Auxiliary Electrical Equipment Room Ventilation System

The auxiliary electrical equipment room houses equipment and systems essential for safe shutdown, including the reactor protection system motor-generators and instrumentation, the engineered safety system generators, and essential relays and switchgear. The station SWS supplies the ventilation system for this area. Portions of the SWS are located in a non-tornado-missile-protected section of the crib house.

It is the staff's position that the licensee provide protection of the SWS or demonstrate that the necessary equipment located in the auxiliary electrical equipment room is adequately ventilated.

By letter dated December 6, 1982, the licensee committed to handle the auxiliary electrical equipment room ventilation system upgrade as part of the TMI Action Plan Item III.D.3.4, "Control Room Habitability." Therefore, backfitting is not required.

4.5.2 Station Battery Systems

The staff's safety evaluation forwarded by letter, dated June 28, 1982, stated that the station batteries are not protected from tornado missiles because they are located in a room with concrete block walls. During the August 1982 site visit, the integrated assessment team observed the battery room area. Although the batteries themselves are contained in a concrete block wall enclosure, that enclosure is located within the east turbine building, which has reinforced concrete walls at least 12 in. thick. Therefore, the staff concludes that the batteries are adequately protected from tornado missiles and backfitting is not required.

4.5.3 Diesel Generator Ventilation

The diesel generator air intake and exhaust systems are not protected from tornado missiles. Damage to the intake or exhaust system could result in diesel generator failure.

(1) <u>Diesel Generator 2 (DG 2)</u>

The DG 2 air intake and exhaust systems are located on the main floor of the turbine building above the tornado-protected area. Loss of DG 2 air intake would not result in loss of function because air can be drawn from inside the turbine building below the main floor. However, loss of the DG 2 exhaust could result in loss of function if the exhaust were to fill the DG intake area and thus result in choking of the DG. It is the staff's position that the licensee must provide assurance that the DG will remain operable in the event that the exhaust system is damaged.

(2) <u>Diesel Generator 2/3 (DG 2/3)</u>

DG 2/3 is located in a separate reinforced concrete structure south of the Unit 3 reactor building. The air intake and exhaust units are located on the roof of that building and are not protected from tornado missiles. Loss of either air intake or exhaust could result in loss of DG 2/3 caused by choking from lack of air or inundation with exhaust fumes. Therefore, it is the staff's position that the licensee must provide assurance that DG 2/3 will remain operable if the ventilation systems are lost.

4.5.4 Exterior Tanks

During the August 1981 site visit, the staff identified the condensate storage tanks (CSTs) as external tanks and thus not protected from tornado missiles. Because the CSTs are used in various scenarios for safe shutdown, it is the staff's position that the CSTs should be protected from tornado missiles or the licensee should provide assurance that safe shutdown can be accomplished using missile-protected systems or components.

4.6 Topic III-4.B, Turbine Missiles

10 CFR 50 (GDC 4), as implemented by Regulatory Guide 1.115 and SRP Section 3.5.1.3, requires that structures, systems, and components important to safety be appropriately protected against dynamic effects, which include potential missiles.

The safety objective of this review is to ensure that all the structures, systems, and components important to safety (identified in Regulatory Guide 1.117) have adequate protection against potential turbine missiles either because of structural barriers or a high degree of assurance that failures at design or destructive overspeed will not occur.

General Electric is currently analyzing the probability for generating turbine missiles generically for its turbine designs. This analysis will consider material properties, turbine disc design, inservice inspection intervals, and overspeed protection system characteristics as they relate to destructive overspeed missile generation. The results of this analysis will be submitted to the staff and will identify recommended inspection intervals for the disc and control valves based on plant-specific turbine characteristics and test results.

To achieve the objective of unlikely failures in the interim, the staff recommends that

(1) volumetric inspections of low-pressure turbine discs be conducted in accordance with General Electric procedures during the next refueling outage unless the discs have been volumetrically inspected within the past 3 years

(2) an inservice inspection program be developed and implemented that requires turbine disassembly at approximately 3-year intervals and inspection of all normally inaccessible parts performed in accordance with the procedures suggested by the turbine manufacturer

(3) all main steam stop and control valves and reheat stop and intercept valves be dismantled and inspected at approximately 3-year intervals

(4) main steam stop and control valves and reheat stop and intercept valves be exercised at least once a week by closing each valve fully.

Dresden Unit 2 does not comply with Item (2) of the staff's recommended inspection program. The licensee's current inspection schedule calls for a complete reinspection of the turbines within 6 years of the last inspections. This inspection interval is based on the turbine manufacturer's calculation of crack growth following a January 1981 wheel bore ultrasonic examination.

By letter dated October 8, 1982, the licensee has provided the staff with the proposed inspection schedule for the low-pressure portions of the turbine and the basis for the proposed schedule. The staff will review the schedule and basis for inspection to determine interim acceptability until completion of the General Electric Company's probability analysis.

4.7 Topic III-5.A, Effects of Pipe Break on Structures, Systems, and Components Inside Containment

10 CFR 50 (GDC 4), as interpreted by SRP Section 3.6.2, requires, in part, that structures, systems, and components important to safety be appropriately protected against dynamic effects such as pipe whip and discharging fluids. The safety objective for this topic review is to ensure that if a pipe should break inside the containment, the plant could safely shut down without a loss of containment integrity and the break would pose no more severe conditions than those analyzed by the design-basis accidents.

The staff has compared the licensee's proposed evaluation methods presented in his letter dated August 23, 1982 with the criteria and methods currently used for licensing new facilities. In general it was found that the licensee's program, methods of approach, and criteria used for the evaluation are adequate. During the review, the staff identified those areas where the licensee's methodology differs from current criteria. Those areas are discussed below.

By letter dated November 17, 1982(b), the licensee provided his final report on the effects of high-energy line breaks on systems, structures, and components inside containment. The staff is currently reviewing the licensee's report.

4.7.1 Jet Impingement on Target Pipe

In considering the damage criteria, the licensee has used the assumption that a jet or whipping pipe is considered to inflict no damage on other pipes of equal or greater size and equal or greater thickness.

The licensee provided some justifications leading to the conclusion that the same rule that is applicable to pipe whip should also be applicable to jet impingement considerations. However, the staff feels that the energy absorption mechanism for a pipe-to-pipe impact is different from that for jet impingement on a pipe. Therefore, it is the staff's position that the licensee should evaluate and address the effects of jet impingement regardless of the ratio of impinged and postulated broken pipe sizes.

4.7.2 Broken-Pipe Impact on Target Piping

In determining the acceptability of target piping, the licensee has used the criterion that the limiting factor for an applied equivalent static load is that the resulting strain in the target piping material does not exceed 45% of the minimum ultimate uniform strain of the material at the appropriate temperature. This criterion is acceptable for avoiding cascading pipe breaks. However, some piping systems are required to deliver certain rated flow and should be designed to retain dimensional stability when stressed to the allowable limits associated with the emergency and faulted conditions; i.e., the functional capability of the piping is required to be demonstrated. The licensee was requested to provide justification to ensure that the target piping will remain functional as a result of jet impingement and pipe whip interactions.

The licensee indicated that he has performed a parametric study covering a range of geometric and load parameters. The results of the nonlinear finite-element dynamic analysis indicated the coexistence of large localized strain levels and small global deformations. Thus, the licensee determined that it is possible to achieve strain levels approaching 45% of the minimum uniform ultimate strain of the material in a localized region without affecting the overall deformation or functionability of the target piping.

In reviewing the example in the licensee's parametric study submitted in his August 23, 1982 letter, the staff found that the 45% of the minimum uniform ultimate strain reached at the point of load application was a global strain because a beam model was used for analysis.

Therefore, it is the staff's position that the licensee demonstrate that the localized deformation associated with a global strain of 45% of the minimum ultimate uniform strain of the material would not affect the functional capability of the target piping.

4.7.3 Detectability Requirements

The licensee's approach for the alternative safety assessment for selected high energy pipe break locations using fracture mechanics analysis is not consistent with the staff's guidance on the subject. For example, the licensee did not address the detectability requirements necessary to detect through-wall cracks, with a length equal to twice the wall thickness, in piping systems that have minimum flow rates associated with normal operating conditions. The licensee proposed an approach based on the leak-before-break concept consisting of the following steps:

(1) The initial crack size is based on a Code-allowable surface defect.

(2) Crack growth is based on a fatigue mechanism.

(3) The end-of-life crack size reflects the growth potential of the initial crack under expected operating conditions.

(4) The end-of-life crack size is compared with the critical crack length to establish the margin of safety.

(5) If the end-of-life crack becomes a through-the-wall crack, the leakage for this crack length is calculated.

(6) The leakage from a through-the-wall crack that is of critical length is calculated, and the margin of safety on leakage from the critical-length crack as compared with the leakage from the end-of-life crack is established.

(7) For the specific postulated break location, the current capability to detect leakage is determined, and this capability is compared with the leakage from the critical-length crack. Additional leakage detection capability as required to ensure that the margin of safety on leakage detection is greater than 100 percent is provided.

The staff has reviewed the licensee's approach and found it is not justified. The licensee's analysis is based on pipe crack caused by fatigue failure of the pipe. The staff's position is that piping failures are more likely caused by other mechanisms (i.e., stress-corrosion cracking). Therefore, it is the staff's position that the licensee follow the staff's guidance for resolution of unresolved interactions.

4.7.4 Criteria Implementation

In the course of the staff's review, two areas were identified where the licensee's approach was found generally acceptable pending staff review of the analysis results. These areas are

(1) pipe whip load formulation
(2) interaction of pipe whip and jet impingement with the containment liner

Therefore, to complete its evaluation of these items, the staff requires the licensee to provide the criteria and results for pipe whip load formulation.

By letter dated August 31, 1982, the Jersey Central Power & Light Company described an evaluation that was performed for the Oyster Creek Nuclear Generating Station. This report supported a conclusion that (1) the interaction between the drywell liner and a whipping pipe could only be glancing blow; (2) no sharp edges could hit the liner in a penetrating direction; and (3) the liner displacement is limited by the concrete drywell which will prevent any rupture of the liner by a recirculation line, main steam line, or feedwater piping. The staff has reviewed the Oyster Creek submittal with regard to its applicability to Dresden Unit 2. Because of the similarity in design, the staff has found that the Oyster Creek results are, in general, applicable to Dresden Unit 2. Therefore, it is the staff's position that the licensee review the Oyster Creek evaluation to ensure that the criteria and methodology used for both pipe whip and jet impingement are applicable to the Dresden Unit 2 design.

4.8 Topic III-5.B, Pipe Break Outside Containment

10 CFR 50 (GDC 4), as implemented by SRP Sections 3.6.1 and 3.6.2 and Branch Technical Positions (BTP) MEB 3-1 and ASB 3-1, requires, in part, that structures, systems, and components important to safety be designed to accommodate the dynamic effects of postulated pipe ruptures. The safety objective for this topic review is to ensure that if a pipe should break outside the containment, the plant can be safely shut down without a loss of containment integrity.

The effects of pipe breaks for both the main steam isolation condenser and reactor water cleanup lines between the containment penetration area and the isolation valve outside containment with an assumed single active failure of the inboard containment isolation valve would result in a loss-of-coolant accident (LOCA) outside containment. Current licensing criteria for this event ensure that a pipe break between the outside isolation valve and the containment wall is unlikely. This is accomplished by ensuring low pipe stress (BTP MEB 3-1) and high-quality pipe (i.e., seismic Category I).

No stress data are available to demonstrate that these piping systems between the containment penetration and the isolation valve outside containment meet the stress limits of BTP MEB 3-1. A limited risk assessment of the importance of the various postulated pipe breaks as unisolable LOCAs was conducted for Dresden Unit 2. It was determined that the LOCA frequencies associated with these pipe breaks are all less than 2 x 10^{-7} per year, since both a pipe break outside containment and a failure of an isolation valve inside containment are necessary for this sequence. Even if all these events led to core melt with release, the higher frequencies of other core-melt sequences coupled with the virtual certainty of containment failure after core melt makes these LOCAs negligible from a risk perspective. In addition, the small frequencies of pipe breaks result in a similar conclusion regarding the physical effects associated with the pipe break. Therefore, the probabilistic risk assessment (PRA) rated the importance to risk of pipe breaks between the containment penetration and the isolation valve outside containment as low.

Backfitting, therefore, is not required.

4.9 Topic III-6, Seismic Design Considerations

10 CFR 50 (GDC 2), as implemented by SRP Sections 2.5, 3.7, 3.8, 3.9, and 3.10 and SEP review criteria (NUREG/CR-0098, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants"), requires that structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes.

4.9.1 Piping Systems

The review of the adequacy of all safety-related piping supports is being reviewed for compliance with Office of Inspection and Enforcement (IE) Bulletins 79-02 and 79-14 and was not duplicated as a part of the SEP piping seismic audit analyses. As discussed in Chapter 6 of NUREG/CR-0891, "Seismic Review of Dresden Nuclear Power Station - Unit 2 for the Systematic Evaluation Program," the methods used for the original seismic piping design, especially for the design of piping supports, may not have been conservative. In response to NRC's "Notice of Violation," the licensee reported that a significant number of modifications to existing supports and the addition of new supports would be required to complete his effort in response to IE Bulletin 79-14. The licensee has proposed to correct, on a priority basis, piping support deficiencies associated with (1) the reactor coolant system pressure boundary piping up to the first normally closed isolation valve or the first isolation valve capable of being closed and (2) piping necessary to ensure at least one path for reactor decay heat removal. The licensee has proposed to complete all work associated with IE Bulletin 79-14 by December 31, 1983.

The Final Safety Analysis Report (FSAR) seismic input (Housner ground-response spectrum anchored at 0.2 g) is more conservative than the site-specific spectrum that was developed by NRC for seismic reevaluation of the Dresden site. On the basis of the low probability of an earthquake with ground motion that exceeds the NRC's site-specific spectrum, the conservatism of the FSAR seismic input, and the margins that exist in FSAR design criteria for piping systems, the staff has determined that it is appropriate to resolve the "adequacy" or conservatism of the original design of piping supports as part of the IE Bulletin 79-14 effort.

4.9.2 Mechanical Equipment

During the audit review of mechanical and electrical equipment, the Senior Seismic Review Team found that information was lacking for three of the nine mechanical equipment items sampled. These items are discussed below.

(1) The staff identified a lack of information with regard to motor-operated valves (MOVs). By letters dated July 7, 1982, September 3, 1982, and November 3, 1982, the licensee provided additional information regarding pipe stress resulting from MOVs. The staff is currently reviewing this information.

(2) The staff lacks sufficient information to evaluate the structural integrity of the reactor vessel and internal supports. The staff will review the analysis of the Oyster Creek reactor vessel and internal supports to determine its applicability to the Dresden 2 design. This information will be used to evaluate the capability of the reactor vessel and internal supports to withstand the SEP-defined earthquake.

(3) The staff lacks sufficient information to evaluate the structural integrity of the recirculation pump and supports. Therefore, the staff will require the licensee to provide further information regarding the capability of the recirculation pump and supports to withstand the SEP-defined earthquake without loss of structural integrity.

4.9.3 Qualification of Cable Trays

The staff lacks sufficient information to conclude that the seismic qualification of electrical cable trays is acceptable. A program was undertaken by the SEP Owners Group intended to provide a set of general analytical methodologies for the seismic qualification of cable trays; this program has not been completed. It is the staff's position that the licensee implement a plant-specific analysis of the structural integrity of cable trays on completion of the SEP Owners program and, if necessary, upgrade cable tray support systems to ensure their ability to maintain their integrity under safe shutdown earthquake loading.

4.9.4 Ability of Safety-Related Equipment To Function

The NRC has initiated a generic program to develop criteria for the seismic qualification of equipment in operating plants as an unresolved safety issue (USI A-46). Under this program, an explicit set of guidelines (or criteria) that should be used to judge the adequacy of the seismic qualification (both functional capability and structural integrity) of safety-related mechanical and electrical equipment at all operating plants will be developed.

4.10 Topic III-7.B, Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria

10 CFR 50 (GDC 1, 2, and 4), as implemented by SRP Section 3.8, requires that structures, systems, and components be designed for the loading that will be imposed on them and that they conform to applicable codes and standards.

Code, load, and load combination changes affecting specific types of structural elements have been identified where existing safety margins in structures are significantly reduced from those that would be required by current versions of the applicable codes and standards. Approximately 30 specific areas of design code changes potentially applicable to the Dresden Unit 2 plant have been identified for which the current code requires substantially greater safety margins than did the earlier version of the code, or for which no original code provision existed.

The significance of the identified code changes cannot be assessed until a plant-specific review of their applicability, as well as of margins in the original design, is completed. This does not infer that existing structures have inadequate safety margins. The review, however, will clarify if the original margins are comparable to those currently specified. This will include consideration of the appropriate applied loads (e.g., roof loading resulting from probable maximum precipitation and snow) and load combinations.

By letter dated August 2, 1982, the licensee provided information regarding the applicability of the identified code changes to the Dresden Unit 2 plant and an assessment of the as-built safety margins. This information is currently being reviewed by the staff and will be addressed in a supplement to this report. Any further actions required of the licensee will be identified following staff review of the consultant's report.

4.11 Topic III-8.A, Loose-Parts Monitoring and Core Barrel Vibration Monitoring

10 CFR 50 (GDC 13), as implemented by Regulatory Guide 1.133, Revision 1, and SRP Section 4.4, requires a loose-parts monitoring program for the primary system of light-water-cooled reactors. Dresden Unit 2 does not have a loose-parts monitoring program that meets the criteria of Regulatory Guide 1.133.

A loose-parts monitoring program could provide an early detection of loose parts in the primary system that could help prevent damage to the primary system. Such damage relates primarily to

(1) damage to fuel cladding resulting from reheating or mechanical penetration

(2) jamming of control rods

(3) possible degradation of the component that is the source of the loose part to such a level that it cannot properly perform its safety-related function

Backfitting of a loose-parts monitoring program is being considered in Revision 1 to Regulatory Guide 1.133. If the staff decides to implement the recommendations of this revision, then the need to implement a loose-parts monitoring program on operating reactors will be addressed generically.

The following factors were considered in making the recommendation that no backfitting be done at this time:

(1) A summary of 31 representative loose-parts incidents at 31 reactors (from the value-impact statement of Revision 1 to Regulatory Guide 1.133) indicates that structural damage occurred as a result of loose parts in only nine incidents. None of these incidents caused a safety-related accident.

(2) Most loose parts can be detected during refueling inspections.

(3) The limited PRA of this issue for Dresden Unit 2 concluded that eliminating loose parts-induced transients by installing a loose-parts monitoring system would have no effect on risk.

Backfitting, therefore, is not recommended for Dresden Unit 2 at this time.

4.12 Topic III-10.A, Thermal-Overload Protection for Motors of Motor-Operated Valves

10 CFR 50.55a(h), as implemented by Institute of Electrical and Electronics Engineers (IEEE) Std. 279-1971 and 10 CFR 50 (GDC 13, 21, 22, 23, and 29), requires that protective actions be reliable and precise and that they satisfy the single-failure criterion using quality components. Regulatory Guide 1.106 presents the staff position on how thermal-overload protection devices can be made to meet these requirements.

The Dresden Unit 2 design does not meet current licensing criteria for all safety-related valve functions because the adequacy of the setpoints for unbypassed thermal overloads has not been established.

4.12.1 Thermal Overloads

Because poor valve reliability may lead to the failure of more than one valve during emergency conditions and multiple valve failures have not been analyzed for their effect on system performance and plant safety, the staff recommends that action should be taken to improve valve reliability.

The limited PRA of this issue for Dresden Unit 2 concluded that a single valve can have its unavailability reduced by about 14% by the elimination of spurious thermal overload trips by bypassing the thermal overload protection. It was concluded that because many valves are affected, the issue is of medium importance to risk.

It is the staff's position that the licensee either bypass thermal overloads with an emergency core cooling system signal or ensure the adequacy of the setpoints for unbypassed thermal overloads.

By letter dated December 6, 1982, the licensee committed to provide an evaluation to show the adequacy of setpoints for unbypassed thermal overloads.

4.12.2 Torque Switches

In MOV designs that use a torque switch instead of a limit switch to limit the opening or closing of the valve, the automatic opening or closing signal should be used in conjunction with a corresponding limit switch, and thermal overload switches should remain as backup protection over the first 10% of valve travel. The licensee has investigated the plant design and has reported that a limit switch bypasses the torque switch to initiate valve movement in all cases. Thus, the staff considers this issue resolved, and backfitting is not required.

4.13 Topic V-5, Reactor Coolant Pressure Boundary (RCPB) Leakage Detection

10 CFR 50 (GDC 30), as implemented by Regulatory Guide 1.45 and SRP Section 5.2.5, prescribes the types and sensitivity of systems and their seismic, indication, and testability criteria necessary to detect leakage of primary reactor coolant to the containment or to other interconnected systems. Regulatory Guide 1.45 recommends that at least three separate leak detection systems

be installed in a nuclear power plant to detect unidentified leakage of 1 gpm from the RCPB to the primary containment within 1 hour. Leakage from identified sources must be isolated so that the flow of this leakage may be monitored separately from unidentified leakage. The detection systems should be capable of performing their functions after certain seismic events and of being checked in the control room. Of the three separate leak detection methods recommended, two of the methods should be (1) sump level and flow monitoring and (2) airborne particulate radioactivity monitoring. The third method may be either monitoring the condensate flow rate from air coolers or monitoring airborne gaseous radioactivity. Other detection methods--such as monitoring humidity, temperature, or pressure--should be considered to be indirect indications of leakage to the containment. In addition, provisions should be made to monitor systems that interface with the RCPB for signs of intersystem leakage by methods such as monitoring radioactivity and water levels or flow.

A limited risk assessment of the importance of the sensitivity of leakage detection systems to risk was conducted for Dresden Unit 2 by using the Millstone Unit 1 Integrated Reliability Evaluation Program (IREP) study. This study only addressed leakage detection as it related to the small-break LOCA (as described in Appendix D). For this event, it was determined that the importance of leak detection capability (i.e., the sensitivity of detectors to leak rate and time) to risk was very dependent on time for a leak to become a break. If the leak-before-break time was short (less than 1 hour) or long (more than 8 hours), the benefits of improved leak detection capability were low. This occurs because the existing Dresden systems can detect leak rates of 1 gpm in about 8 hours, and current criteria would require detection of a leak of 1 gpm in 1 hour. Further, the LOCA sequence for boiling-water reactors (BWRs) is not a dominant sequence. Therefore, the PRA rated the importance of increased sensitivity of leakage detection systems to risk as low. However, this assessment does not address the staff's principal concern with respect to leakage detection, which is not the LOCA event but is related to high energy pipe break (HEPB) discussed in Section 4.7.

For some postulated break locations, where separation and/or restraint is not practical or possible to mitigate the effects of an HEPB, it may be necessary to use local leak detection. The current licensing position of detection of a leak of 1 gpm within 1 hour may not be sufficient for consideration of HEPBs.

It is the staff's position that leakage detection systems and sensitivity should be reviewed in conjunction with "Effects of Pipe Breaks on Structures, Systems, and Components Inside Containment" (Topic III-5.A), and that the limited PRA based on the LOCA for BWRs does not reflect the staff's principal concern with respect to RCPB leakage.

4.13.1 System Sensitivity

The sump pump actuations are not monitored continuously as recommended in Regulatory Guide 1.45. However, the equipment and floor drain sumps in the drywell are pumped at the beginning of each shift. The amount is recorded and compared with that recorded during previous shifts to determine changes or trends. Since the accident at TMI, the operator, on receiving a sump high level alarm, manually initiates the pumping process. Again, the amount is recorded and compared to determine changes or trends.

Although leak rate trends are determined once per shift, this does not meet the current requirements for being able to detect a 1-gpm leak rate in 1 hour.

In addition, the other monitoring systems (airborne particulate and gaseous radioactivity) do not meet the staff's requirements for system sensitivity. It is the staff's position that the necessary sensitivity of leakage detection systems be determined in conjunction with the resolution of SEP Topic III-5.A regarding pipe breaks inside containment. The adequacy of the existing leakage detection systems will be determined once the required system sensitivity is determined.

The licensee has submitted his final report for Topic III-5.A. The staff is currently reviewing that report.

4.13.2 Seismic Qualification

The detection systems do not meet the recommendations of Regulatory Guide 1.45 with regard to the staff's position that the leakage detection systems be operable following a seismic event. It is the staff's position that the leakage detection system should be qualified to a safe shutdown earthquake seismic event or appropriate procedures should be available to specify actions to be taken following seismic events and failure of the leakage detection equipment (i.e., plant shutdown).

4.13.3 System Testability

Both the airborne particulate and gaseous monitoring systems can be tested during normal operation. However, the sump level monitoring system does not meet the recommendations of Regulatory Guide 1.45 with regard to testability during plant operation. The current practice of pumping the sump and recording the amounts every shift ensures sump pump and level monitoring operability. Therefore, the staff concludes that current operating practice meets the intent of the system testability requirements. Therefore, backfitting is not required.

4.14 Topic V-6, Reactor Vessel Integrity

Appendices G and H to 10 CFR 50 and 10 CFR 50.55a(g), as implemented through Regulatory Guide 1.99, require that reactor vessel integrity be ensured by review of aspects such as fracture toughness, surveillance programs, and neutron irradiation.

The staff's review of this topic identified the following issues:

(1) The licensee was asked to supply information on specific reactor vessel materials and surveillance materials.

(2) At the next surveillance capsule test, the licensee should determine the upper shelf energy.

By letter dated March 31, 1982, the licensee asked to amend the Dresden Unit 2 Technical Specifications as they pertain to Appendices G and H to 10 CFR 50. As part of its review of the proposed amendment, the staff will address and resolve Items (1) and (2). Because this evaluation is being performed as part of routine licensing actions of operating reactors, no further action is required for this topic.

4.15 Topic V-10.B, Residual Heat Removal System Reliability

The topic review for SEP Topic V-10.B was performed in conjunction with Topics V-11.B, "Residual Heat Removal System Interlock Requirements" and VII-3, "Systems Required for Safe Shutdown." The differences identified for these topics will be discussed in Section 4.25, which addresses Topic VII-3.

4.16 Topic V-11.A, Requirements for Isolation of High- and Low-Pressure Systems

10 CFR 50.55a, as implemented by SRP Section 7.6 and BTP ICSB 3, requires that interlock systems important to safety be adequately designed to ensure their availability in the event of an accident. This includes those systems with direct interface with the reactor coolant system that have design pressure ratings lower than the reactor coolant system design pressure.

The reactor water cleanup (RWCU) system does not satisfy the current licensing requirements. Isolation on the suction side of the RWCU system is provided by three motor-operated valves (MOVs), an inboard valve (closest to the reactor coolant system), a pump suction valve, and a pump bypass valve. Isolation on the discharge side is provided by an MOV and three check valves. All the MOVs have position indication in the control room. None of the MOVs will open if pressure in the low-pressure portions of the system is higher than its design pressure. All the MOVs will close on high RWCU system temperature, low flow, high RWCU system pressure, low reactor level, high drywell pressure, or loss of control power. The interlocks for these valves use the same sensors and relays. Because the interlocks for the isolation valves are not independent (i.e., one pressure sensor actuates all three valves), the staff has determined that Dresden Unit 2 does not comply with current licensing requirements.

By letter dated September 15, 1982, the licensee provided information regarding the relief capacity of the RWCU system so that pressure is maintained within the design limits assuming failure of the valve pressure switch.

The scenario of concern is a failure in the full-open position of the RWCU system pressure control valve. The original General Electric design specification for the RWCU system states, "Relief valve sizing must ensure system integrity. In sizing the relief downstream of the PRV maximum relief flow conditions must be used." The maximum flow through the pressure control valve in the failed-open position is 1,300 gpm. Normal RWCU system flow is approximately 650 gpm. Pressure relief is provided by two downstream relief valves, a 6-in. relief valve that is rated at 1,260 gpm at 150 psig, and a 1-in. relief valve that is rated at 40 gpm at 140 psig. The 6-in. relief valve discharges to the main condenser. The 1-in. relief valve discharges to the reactor building equipment drain tank. This tank has a 5,000-gal capacity and is automatically pumped on high level to the 33,000-gal waste collector tank at a rate of 50 gpm. Therefore, adequate relief capacity is provided to handle full system flow assuming a pressure control valve failure. Actuation of the relief valve, assuming failure of the pressure control valve, results in a loss of reactor coolant to the hotwell.

The operator can detect high pressure in the RWCU system by a high-pressure alarm set at 150-psig or by a high-temperature alarm that monitors the discharge side of the 6-in. relief valve. Both annunciator procedures for these alarms instruct the operator to check for pressure control valve malfunction. The procedure for the high-pressure alarm annunciation also indicates that the system should isolate automatically. The operator has sufficient indication to manually isolate the system if necessary.

By letter dated October 19, 1982, the licensee provided additional information regarding this scenario. The diversion of system flow from the secondary side of the regenerative heat exchangers caused by relief value actuation will result in exceeding the temperature setpoint of the system. A high temperature signal will then initiate a control room alarm and automatic isolation of the RWCU system.

The limited PRA performed for this issue has concluded that the importance to risk depends on proper sizing of the relief capacity of the RWCU system. As described above, the RWCU system's relief capacity is sufficient to handle full system flow, assuming failure of the pressure control valves. Therefore, the PRA classifies the issue's importance to risk as low.

On the basis of the above considerations, the staff has concluded that the design of the RWCU system is adequate to prevent overpressurization and resulting LOCA outside containment. Therefore, backfitting is not required.

4.17 Topic V-II.B, Residual Heat Removal System Interlock Requirements

The review for SEP Topic V-11.B was performed in conjunction with that of Topics V-10.B, "Residual Heat Removal System Reliability," and VII-3, "Systems Required for Safe Shutdown." The differences identified for these topics will be discussed in Section 4.25, as part of Topic VII-3.

4.18 Topic VI-4, Containment Isolation System

10 CFR 50 (GDC 54, 55, 56, and 57), as implemented by SRP Section 6.2.4 and Regulatory Guides 1.11 and 1.141, requires isolation provisions for the lines penetrating the primary containment to maintain an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment. The staff's review of the containment penetrations has identified several areas that do not conform to current licensing criteria for containment isolation.

The limited PRA for Dresden Unit 2 evaluated the contribution to risk of containment isolation. The PRA has concluded that the overwhelmingly dominant portion of risk from nuclear power plants is from core-melt accidents, not other (low-consequence) releases, such as those resulting from non-core-melt accidents, that result in relatively low (compared with core melt) doses to the public. Because of the small size and low design pressure of the Dresden Unit 2 containment, the pressure generated by steam and noncondensible gases during a core melt will fail the containment if no other failure mechanism occurs first. Therefore, because of the characteristics and relative consequences of leakage releases and containment ruptures by overpressure, the PRA has concluded that no benefit can be achieved by increasing the reliability of isolation of the containment because it will fail by overpressure anyway, and thus classifies these issues' importance to risk as low.

On the basis of the PRA analysis, the staff has not recommended physical modifications of the Dresden Unit 2 facility to comply with the GDC requirements. However, because of the significant contribution to offsite radiological consequences from containment leakage following non-core-melt accidents, the staff has recommended modifications in various areas (such as administrative controls), as discussed in the following sections.

4.18.1 Locked-Closed Valves

All valves located between the inboard and outboard containment isolation valves or before the final outboard isolation valve (if there are none inside containment) should be locked closed to ensure the integrity of piping between these valves.

The licensee does use methods of administrative control on many of these lines in the form of valve checklists and outage cards. However, these procedures do not meet current licensing requirements for ensuring that the valves are not inadvertently opened during periods when containment integrity is required. It is the staff's position that these valves should be administratively controlled and locked in a closed position as required by NRC regulations (GDC 55, 56, and 57). It is also the staff's experience that system lineup procedures and valve checklists are not designed to ensure containment integrity; rather they are designed to ensure proper system function. A specific administrative procedure to periodically ensure that containment isolation valves are in the proper position is essential. At other plants manual valves have been left open for extended periods.

The affected valves are located on either test, vent, drain, or capped branch lines that connect to piping penetrating the containment. The valves, which should have mechanical locking devices and for which appropriate administrative control should be provided, are listed in Table II of the staff's safety evaluation report forwarded by letter dated September 24, 1982.

By letter dated November 18, 1982, the licensee committed to review all containment penetrations in the plant and not limit the scope to those provided in the staff evaluation. Because some of the valves are inaccessible during normal operation, this review will be completed and all associated procedural changes will be initiated during the refueling outage scheduled to start on January 8, 1983.

4.18.2 Leakage Detection

The low-pressure coolant injection (LPCI), core spray, and reactor building closed cooling water (RBCCW) systems are closed systems as defined in GDC 57; they are provided with remote manual isolation valves rather than automatic isolation valves.

During the Appendix J leak detection review, the staff identified the RBCCW system valves 3702 and 3703 as containment isolation valves requiring leakage detection capability. By letter dated August 27, 1982, the licensee committed to install the proper leak rate test taps on the RBCCW lines. The modifications are expected to be completed during the fall 1984 refueling outage.

The other identified systems serve an essential emergency core cooling system function and the staff agrees that automatic isolation valves should not be used. However, because operator action is required to initiate isolation, if necessary, the operator must know when to do so. This requires leakage detection capability and appropriate procedures to indicate under what conditions these valves should be closed. The operating station for these remotely operated valves must be accessible, but it need not be in the control room. It is the staff's position that adequate leakage detection and appropriate procedures for operator action should be provided and the operating station should be relocated to an accessible area, where necessary, for the following valves:

(1) LPCI
 1501-5A, B, C, D
 1501-22, A, B

(2) Core spray
 1402-3A, B
 1402-25A, B

By letter dated November 18, 1982, the licensee provided information regarding the ability to detect leakage and appropriate procedures. The staff reviewed the information and has determined that the licensee has the ability to detect leaks through use of sump level monitors. However, the licensee has not provided information regarding the location of the valve operators for all the valves. Furthermore, the staff has determined that the procedures do not address the staff concerns, i.e., actions and precautions to be taken in the event of a system leak during accident conditions when the leakage may contain high levels of radioactivity. Therefore, it is the staff's position that the licensee should provide the location of the motor operators for valves 1501-22A, B and 1402-25A, B and modify the plant emergency procedures to address the staff's concerns.

4.18.3 Manual Isolation Valves

GDC 55, 56, and 57 (Appendix A to 10 CFR 50) state that containment isolation valves should be either automatic or locked closed unless they can be demonstrated acceptable on another defined basis. The staff has identified valves 4327-500, -502, and 1916-500 on the demineralized water supply lines and valve 4609-501 on the service air supply line as manual valves that are not locked closed. (These valves are primary containment isolation valves as opposed to test, vent, or drain lines discussed in Section 4.18.1.)

By letter dated Nobember 18, 1982, the licensee has committed to lock the valves closed and to change the appropriate procedures.

4.18.4 Check Valves as Isolation Valves

The following systems use check valves in series instead of a check valve inside and a remote manual valve outside the drywell for containment isolation as required by GDC 55 and 56. These systems are the feedwater system (valves 2-220-58A and B inside and 2-220-62A and B outside containment) and the high-pressure coolant injection system (HPCI)(valves 2301-34, -45, -71, and -74, all located outside containment). HPCI valves 2301-71 and -74 are actually stop check valves that are locked open and only closed for performance of leak testing; this is equivalent to the use of two check valves outside containment.

The feedwater system supplies the reactor through two parallel 18-in. lines, each containing two check valves in series (one inside and one outside containment). Remote manual isolation valves exist (in the turbine building) at the discharge end of each high-pressure heater stage (three units in parallel).

For the following reasons, replacing a feedwater check valve with a remote manual isolation valve or adding a remote manual isolation valve outside containment is not recommended:

(1) The high-pressure heater discharge valves provide backup isolation capability.

(2) The existing feedwater check valves are subject to local leakage rate tests, in accordance with 10 CFR 50, Appendix J.

(3) The isolation reliability would not be significantly improved by adding a remote manual valve.

Replacing an HPCI stop check or check valve with a remote manual valve or adding a remote manual valve is not recommended for the following reasons:

(1) The existing stop check and check valves are subject to local leak rate testing in accordance with 10 CFR 50, Appendix J.

(2) Two of the valves in question (2301-45 and -74) are on exhaust steam lines from the HPCI turbine discharging to the suppression pool water. This system is a closed system outside containment and a single isolation valve is acceptable.

(3) The remaining valves (2301-34 and -71) return water from the HPCI turbine moisture drain pot to the torus above the water. These are small (2-in.) lines designed to eliminate water slug buildup thereby permitting rapid start of the unit. The system is connected to the standby gas treatment system (SBGTS). Backleakage of radioactivity would be treated by the SBGTS before being discharged to the atmosphere. Therefore, backleakage is not a safety concern.

(4) The isolation capability of either system would not be significantly improved by adding a remote manual valve in place of a check value.

4.18.5 Valve Location

The HPCI condensate drain and turbine exhaust lines contain two check valves outside containment rather than one valve inside and one outside as required by GDC 55. The relative benefit of one valve inside and one valve outside rather than two valves outside containment was evaluated for the Palisades Plant (see NUREG-0820, Appendix D). In this study no improvement could be identified for moving a valve inside containment. This is because the probability of failure of both valves was greater than the probability of failure of the pipe between the containment and the first isolation valve. Because of minimum improvement in containment isolation capability and low importance of leakage to overall BWR risk, backfitting is not recommended. The use of check valves as isolation valves is discussed in Section 4.18.4.

4.18.6 Branch Lines With Single Isolation Valves

The staff has identified branch lines that contain a single locked-closed isolation valve and a threaded capped stop. The single valves are 1501-70 A and B and 1599-27 A and B located on the LPCI system pump suction lines and 1402-10 A and B located on the core spray system pump suction lines. Because of the safety significance of the torus water, it is the staff's position that a threaded cap does not constitute an acceptable isolation barrier because it can be easily removed and is not subject to leak testing.

By letter dated November 18, 1982, the licensee has committed to provide a second isolation valve on each line and to lock the valve closed.

4.19 Topic VI-6, Containment Leak Testing

10 CFR 50, Appendix J, requires that tests be performed to ensure that leakage through the primary reactor containment and systems and components penetrating primary containment shall not exceed allowable leakage rate values as specified in the Technical Specifications or associated bases.

At Dresden Unit 2, the licensee requested exemptions from certain requirements of the containment leak tests. The staff has granted the requested exemptions with the exception of (1) the reactor building closed cooling water (RBCCW) supply and return isolation valves and (2) the containment airlock.

By letter dated August 27, 1982, the licensee provided a schedule for installation of the proper local leak rate test taps on the RBCCW lines. The current schedule (taking into consideration engineering, procurement, and installation) calls for completion of the modifications during the fall 1984 refueling outage.

4.20 Topic VI-7.A.4, Core Spray Nozzle Effectiveness

10 CFR 50.46 requires that each BWR be provided with an emergency core cooling system designed to provide adequate cooling of the nuclear fuel under postulated accident conditions. Appendix K to 10 CFR 50, "ECCS Evaluation Models," sets forth the required and acceptable factors of the evaluation models.

Information derived from Japanese core spray tests suggested that the central fuel bundles of a BWR/3 core may receive low core spray flow. Dresden Unit 2 is a BWR/3 plant.

The staff is reviewing this concern independently of the SEP as a matter related to Generic Issue A-16, "Steam Effects on BWR Core Spray Distribution." The staff has evaluated the related information and has concluded that the Japanese data do not provide a basis for changing its conclusion that core spray flows for a BWR/3 are not less than the minimum flow required for core spray heat transfer.

Therefore, the staff has concluded that no further SEP action is necessary for the following reasons:

(1) The Japanese data for a BWR/5 may be applicable only to a BWR/4 and a BWR/5 because they have a similar spray nozzle design. The BWR/3 spray nozzle design is different from BWR/4 or BWR/5 designs.

(2) Even though there are no core spray test data in a steam condition for a BWR/3 configuration, a BWR/6 30° sector steam test and 360° full-scale tests in an air environment performed in the United States indicate that the core spray overlaps the center bundles causing high flow rate over the central region of the core. As a result, flow to each bundle is not less than the minimum spray flow required for core spray heat transfer.

(3) General Electric (GE) has informed the staff in a conversation that GE analyses show that for limiting cases of a BWR/3 with core spray assumed to flow down peripheral channels to increase the reflood rate (as observed in the Lynn test), the calculated peak clad temperature did not exceed the 10 CFR 50.46 limit of 2200°F with no credit taken for the spray cooling effect. The staff has requested GE to submit these analyses for its review.

4.21 Topic VI-7.C.1, Appendix K - Electrical Instrumentation and Control (EI&C) Re-Reviews

10 CFR 50 (GDC 2, 4, 17, and 19), as implemented by SRP Sections 8.3.1 and 8.3.2, requires that the onsite electric power distribution system be designed to provide (1) redundancy of safety-related components and systems, (2) electrical independence between redundant portions, and (3) physical separation between redundant components of the system.

Physical separation of power, instrumentation, and control cables associated with the various components (buses, switchgear, etc.) of the distribution systems, as it complies with Regulatory Guide 1.75, is included with this topic review. Additional review guidelines for cable separation, as well as for the physical separation of redundant distribution systems, is defined in 10 CFR 50, Appendix R.

A limited PRA for this issue has been conducted. The items discussed in Sections 4.21.1 and 4.21.5 have not been evaluated because it is an assumption of PRA that the equipment is of an adequate design to perform the function for which it is intended. The remaining items were analyzed with respect to evaluating the potential impact of the staff's recommended actions on plant safety. On the basis of discussions with plant personnel, the PRA contractor assumed that procedures already exist that conform to the staff's recommendations and that implementation of the staff's recommendations would not increase plant reliability. Thus, the PRA has classified this issue's importance to risk as low.

The staff has reviewed existing plant procedures and has found they do not conform to its recommendations. Therefore, the conclusions reached in the PRA are invalid. It is the staff's position that the licensee take appropriate action, as described in the following sections, to ensure that the plant EI&C features will perform their intended safety functions.

4.21.1 Breaker Adequacy

(1) Battery Charter Isolation

Division I motor control centers (MCCs) and Division II MCCs can be subjected to common faults and transients that may occur on the dc system if their respective battery charger output breakers are both closed. Individual manually operated circuit breakers connect the outputs of each of the chargers to the single battery and dc loads. The manual aspect of the design meets review guidelines defined in Position D.4.c of Regulatory Guide 1.6. However, there are no interlocks to prevent the simultaneous closing of the manually operated circuit breakers. This is a deviation from review guidelines defined in Position D.4.d of Regulatory Guide 1.6. However, the two ac breakers and two dc breakers per charger plus the isolation characteristics of each charger provide isolation and separation of ac power sources. Thus, there is no direct connection of ac power systems. It is the staff's position that the licensee verify the adequacy of the protective relaying so that operator error will not result in a loss of redundant ac sources.

(2) <u>125-V DC Automatic Transfer</u>

The design of the 125-V dc system provides for the automatic transfer of the control and instrument power for the diesel generator (DG 2/3) from the Unit 2, Division I, 125-V dc distribution panel to the Unit 3, Division I, 125-V dc distribution panel. The Unit 3 battery/battery-charger combination is the power source for both Unit 2, Division II, and Unit 3, Division I. Therefore, the diesel generator control and instrument loads are automatically transferred between redundant divisions. This is a deviation from current review criteria.

The worst condition would be a fault on the circuit feeding the DG 2/3 load. For a fault at this location to have an effect on the Unit 2 125-V dc system, two breakers would have to fail. For this failure to propagate to the Unit 3 125-V dc system after the Unit 2 125-V dc system has had multiple failures, the load must transfer to Unit 3 and two more breakers would have to fail. Breaker-failure mechanisms exist that may cause failure of one or more ac or dc breakers in series and are not limited to gross mechanical failures. Such mechanisms include

(a) failure of a load breaker to clear a fault before the bus feeder reaches its trip setpoint (may be caused by a lack of adequate protection curves resulting from failure to revise settings as new loads and/or sources are added to a system)

(b) a ground fault tripping an ac feeder breaker instead of a load breaker (caused by inadequate ground-fault protection)

(c) protective relay setpoint drift outside the error band assumed in the coordination of load and feeder breakers (caused by infrequent testing of the relays)

It is the staff's position that the licensee verify the adequacy of the protective relaying so that a fault in the DG 2/3 control system will not result in a loss of redundant dc sources.

(3) <u>Standby 250-V Battery Chargers</u>

The standby 250-V battery charger (a Division I system) is supplied power from either the Unit 3 or Unit 2, Division II, power source through a key-interlock switch. When power is supplied from Division II of Unit 3 to the Unit 2 battery, there is sharing between Units 2 and 3. This sharing is covered by

SEP Topic VI-10.B. When power is supplied from Division II of Unit 2 to the Unit 2 battery, there is an interconnection between redundant divisions. It is the staff's position that the licensee verify the adequacy of the protective relaying so that a fault in one dc system would not be transferred to the other dc system.

By letter dated December 6, 1982, the licensee has committed to provide either a short-circuit analysis or a coordination study for the battery charger isolation, 125-V dc automatic transfer, and 250-V battery chargers.

4.21.2 Disconnect Links

Division I main dc Bus 2 is interconnected to Division II reserve dc Bus 2 through circuit breakers, disconnect links, and the Division I 125-V dc distribution panel. There are 16 locations where similar interconnections can be made between redundant divisions. The breakers are molded case breakers and are not of the type that can be racked out; they can only be placed in the open position. These breakers are only used during maintenance operations. However, no administrative controls are provided to verify that the disconnect links are placed in the open position following completion of the maintenance activities.

By letter dated December 6, 1982, the licensee has committed to address the use of disconnect links in the plant procedures.

4.21.3 Use of Breakers During Power Operations

There is no control circuitry at Dresden Unit 2 that is designed to open the two tie breakers (252-2829 and 252-2928) for redundant 480-V Buses 28 and 29 concurrently with the loss of offsite power. Nor are there limiting conditions for operation in regard to these breakers with respect to how long they may be closed during normal operation. This could result in overload of a diesel generator and is a deviation from review guidelines.

By letter dated December 6, 1982, the licensee has committed to address the use of breakers 252-2829 and 252-2928 during power operation in the plant procedures.

4.21.4 Operation With Failed Battery

As previously described in Section 4.21.1(2), DG 2/3 instrumentation and control power can be connected to the Unit 3 125-V distribution panel. This transfer has raised a concern with the staff regarding the availability of the battery systems. The Technical Specifications for Dresden Unit 2 establish a limiting condition for operation (LCO) of 7 days for continued operation with a battery or diesel generator out of service. This 7-day limit is not in agreement with Standard Technical Specification (STS) limits. The STS limits, which are based on generic risk estimates, require that a failed battery system be restored to operable status within 2 hours or the plant be shut down. Operation with a failed diesel generator is acceptable for 7 days because of the availability of two operable diesel generators and two battery systems. However, a failed battery system would leave only a single battery system operable for sharing between two units. It is the staff's position that continued operation with one operable system is undesirable.

By letter dated December 6, 1982, the licensee has committed to comply with a 2-hour Technical Specification limit for operation with a failed battery.

4.21.5 Isolation of Class 1E Sources From Non-Class 1E Loads

The 480-V ac Switchgear 27 normally receives ac power from Bus 24. The dc control power is, however, from Division I. This is a deviation from review guidelines because 480-V Switchgear 27 is non-Class 1E. It is the staff's position that the licensee should demonstrate, by suitable short-circuit analyses and coordination curves, that all non-Class 1E loads are adequately isolated from Class 1E sources by at least two circuit breakers in series (e.g., Switchgear 27 feeder breaker and individual load feeder breakers should be coordinated to ensure that faults are not transferred to the Class IE bus). Two breakers are physically present; however, their trip devices may not be coordinated.

By letter dated December 6, 1982, the licensee has committed to provide either a short-circuit analysis or a coordination study regarding isolation of Class 1E sources.

4.22 Topic VI-10.A, Testing of Reactor Trip System and Engineered Safety Features, Including Response-Time Testing

10 CFR 50 (GDC 21) requires that the reactor protection system be designed to permit periodic testing of its functioning, including a capability to test channels independently.

10 CFR 50.55a(h), through IEEE Std. 279-1971 and IEEE Std. 338-1971, requires that response-time testing be performed on a periodic basis for plants with construction permits issued after January 1, 1971.

The staff's review of response-time testing at Dresden Unit 2 has shown that mechanical systems that provide the major time delays, such as control rod drive systems, diesel generators, and the major emergency core cooling system valves and pumps and containment isolation valves, are response-time tested. In addition, plant procedures are used to response-time test the reactor protection system logic relays. Only the sensors are not tested. The staff performed a limited PRA of the issue for the Dresden Unit 2 plant to estimate the improvement in overall safety if additional response-time testing was required. The results of this PRA indicated that additional response-time testing has low safety significance. This occurs because this testing is concerned with events on the order of seconds. The IREP studies (Millstone Unit 1, Browns Ferry (NUREG/CR-2802), Arkansas Nuclear One, Calvert Cliffs, and Crystal River Unit 3) have shown that response times of 20 to 40 minutes are sufficient for emergency core cooling system actuation for both BWR and pressurized water reactors. Functional tests are sufficient to demonstrate function on the order of minutes, and these tests are performed at Dresden Unit 2. Therefore, it is the staff's judgment that response-time testing of instrumentation, other than that already required by the Dresden Unit 2 Technical Specifications, should not be required.

4.23 Topic VI-10.B, Shared Engineered Safety Features, Onsite Emergency Power, and Service Systems for Multiple-Unit Facilities

10 CFR 50 (GDC 5) requires that structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their abilitv to perform their safety functions.

4.23.1 Sharing of DC Systems

The 125-V and 250-V dc systems are shared, which is not in compliance with current licensing requirements. However, the staff's review of the present dc designs shows that they satisfy the single-failure criterion if the disconnect links are open and are not paralleled. These devices are to be reevaluated as part of the resolution of Topic VI-7.C.1.

There are no physical or electrical interlocks or LCOs preventing parallel operation of the shared 125-V and 250-V dc battery systems. Such operation, combined with a single failure, would result in a loss of capability to supply accident or safe shutdown loads following a loss of offsite power.

The staff's audit of operating procedures (e.g., DOP 6900-4) indicates that there are no procedures requiring paralleling of the 250-V dc systems during reactor operation. However, there are no requirements to prevent the paralleling of the 250-V batteries.

Plant procedure DOP 6900-6 requires that the 125-V batteries be paralleled as part of the ground-fault detection procedures.

The limited PRA for this issue has determined that the probability of operating the dc batteries in parallel, leading to failure, is very small. Therefore, the PRA classified this issue's importance to risk as low. However, the PRA also identified one aspect of this event that is beyond the scope of the analysis. The possible effects of the fault created in the use of parallelling procedures include: tripping the plant, degradation of the high-pressure coolant injection system, and failure of the isolation condenser system. Different circuit breakers/fuses would have to fail to effect these events. If the worst case is assumed and all three events were to occur, the use of this procedure could lead to a significant accident sequence. Sufficient information is not available to determine the likelihood of this worst case or of the different permutations of faults that may be caused by the use of this procedure to determine their contribution to plant risk.

NUREG-0666 and Regulatory Guide 1.81 establish the basis for the staff's position that dc systems in multiunit nuclear power plants should not be shared. In the case of parallel operation, a single failure could result in a loss of engineered safety features in both plants and, simultaneously, initiate plant transients. Given that a ground fault exists, the wisdom of paralleling 125-V dc batteries (and doubling the available fault current) is questionable. The added possibility of a major upset occuring simultaneously is neither acceptable nor necessary given the availability of other ground-fault detection systems using different techniques.

Therefore, it is the staff's position that the licensee provide assurance that paralleling of the shared 125-V and 250-V dc systems be prohibited during power operation.

By letter dated December 6, 1982, the licensee has stated that he is looking into means of not paralleling the battery systems. Currently, paralleling is the only method for finding high impedance ground faults.

4.23.2 Diesel Generator Bypass

The staff has found that there are no LCOs that require or interlocks that prevent the normal/bypass switches for the DG 2/3 from being in "bypass" during operation of either unit. Such operation, combined with a single failure, could render the required accident and safe shutdown loads inoperable following a loss of offsite power.

In a letter from T. J. Rausch to P. O'Connor dated August 30, 1982, the licensee stated that the operating procedures had been changed so that they require a "normal-normal" alignment of these switches. The staff has reviewed these procedures and found them acceptable.

4.23.3 Battery Status Indication

Complete information of the status of the shared dc batteries, chargers, and buses is not available to operators of each unit. Battery status indication will be addressed under SEP Topic VIII-3.B.

4.23.4 Battery Room Ventilation

The battery room ventilation system is not powered from an onsite source. The staff is concerned because the time of highest hydrogen concentration occurs while the diesel generator is being used to recharge the batteries. The licensee's response was that manual methods could be used to load the vent fan onto DG 2. A review of procedure DGA-12 does not include loading of any fans, although Bus 27 is reenergized. This item is being evaluated as part of SEP Topic IX-5, "Ventilation Systems," and is addressed in Section 4.29.

4.24 Topic VII-1.A, Isolation of Reactor Protection System From Nonsafety Systems, Including Qualifications of Isolation Devices

10 CFR 50.55a(h) through IEEE Std. 279-1971 requires that safety signals be isolated from nonsafety signals and that no credible failure at the output of an isolation device shall prevent the associated protection system channel from meeting the minimum performance requirements specified in the design bases.

4.24.1 Reactor Protection System (RPS) Control Systems

The analog signals from the nuclear flux monitoring system intermediate range monitors (IRMs), local power range monitors (LPRMs), and average power range monitors (APRMs) are not isolated from the control room process recorders and indicating meters as required by IEEE Std. 279.

A limited PRA was performed for this issue. The PRA determined that a fault in the nonsafety part of the nuclear flux monitoring channel or APRM could fail the high neutron flux signal or APRM. However, the probability of reactor protection system (RPS) failure is totally dominated by common-mode mechanical faults associated with the control rod drive system, and eliminating the isolation problem would not effect RPS unavailability. Thus, the PRA classified the issue's importance to risk as low. However, the staff disagrees with the PRA.

The neutron flux monitoring system, consisting of the IRMs, LPRMs, and APRMs, is designed to provide the operator with information required for safe operation of the reactor core and provide inputs to the RPS and rod block circuitry to ensure that power density and level do not exceed preset limits. Because of the safety significance of the neutron flux monitoring systems, it is the staff's position that the licensee provide assurance that common-mode electrical faults occurring in the control room process recorders and indicators will not disable the neutron flux monitoring systems.

By letter dated December 6, 1982, the licensee has committed to verify that the neutron monitoring system is sufficiently isolated from the control room indicators to properly monitor core conditions or install Class 1E isolators.

4.24.2 Process Computer

The APRM scram function is derived from relay actuation resulting from amplified analog signals sensed by these relays. The amplified analog signals are input directly to the process computer with fuses as the isolation device. Fuses do not meet the intent of IEEE Std. 279 for isolation devices (e.g. fuses will not isolate ground faults). It is the staff's position that the licensee should address the adequacy of the isolation circuitry to ensure that the RPS is protected from potential common-mode electrical faults that could be propagated from the process computer.

By letter dated December 6, 1982, the licensee has committed to verify that the neutron monitoring system is sufficiently isolated from the process computer to properly monitor core conditions or install Class 1E isolators.

4.24.3 RPS Channel Power Supplies

Power to the RPS buses is supplied from two motor-generator sets. The isolation of each RPS channel and its motor-generator set does not conform with current licensing criteria as defined in IEEE Std. 308-1974.

The licensee has committed by letter dated December 11, 1980 to install Class 1E protection at the interface between the RPS power supply and the RPS. The licensee has stated that the system will be in accordance with the conceptual design proposed by the General Electric Company and found acceptable by the staff. This modification will be completed during the next scheduled refueling outage.

The staff agrees with the licensee's proposed action.

4.25 Topic VII-3, Systems Required for Safe Shutdown

4.25.1 Procedures for Shutdown From Outside Control Room

10 CFR 50 (GDC 19), as implemented by SRP Section 7.4, requires the capability to promptly achieve and maintain a hot shutdown condition from outside the control room with the potential of capability of achieving subsequent cold shutdown through the use of suitable procedures. During the topic review it was determined that Dresden Unit 2 did not have procedures for accomplishing this objective.

During the August 1982 site visit, the licensee provided the staff with Procedure EPIP 200-20, Revision 1, April 1982, which details the method for achieving and maintaining a hot shutdown condition assuming evacuation of the control room. Instructions are provided for local operation of the isolation condenser, diesel generator start and loading, and operation of the control rod drive and the condensate transfer pumps. This procedure satisfies the staff's position regarding achieving and maintaining a hot shutdown condition from outside the control room. However, the licensee does not have procedures for subsequently achieving a cold shutdown condition.

The licensee's submittal dated July 1, 1982 regarding the Appendix R fire protection review provides a commitment to modify the safe shutdown procedures so that they include the capability to achieve cold shutdown from outside the control room. Since the fire protection reviews are being conducted independently of SEP, no further SEP action is required on this subject. Backfitting, therefore, is not required.

4.25.2 Use of Safety-Grade Systems

10 CFR 50 (GDC 19 and 34), as implemented by SRP Section 5.4.7, BTP RSB 5-1, and Regulatory Guide 1.139, requires that the plant can be taken from normal operating conditions to cold shutdown by using safety-grade systems and either onsite or offsite power, assuming a single failure.

The initial topic review showed that Dresden Unit 2 did not have procedures for achieving cold shutdown from normal operating conditions using only safety-grade systems and either onsite or offsite power, assuming a single failure.

During the August 1982 site visit, the licensee provided the staff with copies of revised procedures (DGP 2-3, Revision 7, May 1981, and DGA 12, Revision 2, June 1981). These procedures provide information regarding various automatic and manual actions to be taken to ensure that the plant can achieve a cold shutdown using the essential safety systems as identified in the staff's topic review. The staff has reviewed the procedures and has concluded that the operating procedure adequately address use of the systems identified as essential to achieve and maintain a cold shutdown condition.

4.25.3 Residual Heat Removal Single-Failure Criteria

10 CFR 50 (GDC 34) requires that a system to remove residual heat be provided with suitable redundancy to ensure that for onsite electric power system operation's the system's safety function can be accomplished, assuming a single

failure. At Dresden Unit 2, long-term cooling is susceptible to single failures if the shared diesel generator is not available to Unit 2.

This problem was addressed in the staff's evaluation of SEP Topic VI-10.B. The staff concluded that Unit 2 shutdown will commence with use of the isolation condenser and the HPCI system until the shared diesel generator can be manually transferred to Unit 2 to support long-term cooling. The staff's audit of operating procedures and drawings has confirmed the adequacy of this method of operation. Backfitting, therefore, is not required.

4.25.4 Inservice Testability

10 CFR 50 (GDC 21), as implemented by IEEE Std. 279-1971, requires that protection systems be designed for inservice testability commensurate with the safety function to be performed.

At Dresden Unit 2, the shutdown cooling system is designed for full reactor pressure but less than full reactor temperature. Therefore, system interlocks are based on temperature requirements. Current licensing criteria for the interlocks are not met because there are no testing requirements.

The limited PRA for this issue has determined that testing of the temperature interlocks would increase the availability of the shutdown cooling system by about 15%. This evaluation is based on the assumption that the temperature interlocks are not tested and that exceeding the design temperature would fail shutdown cooling. Therefore, the PRA classified this issue as having medium importance to risk.

It is the staff's position that the licensee provide for inservice testability of the shutdown cooling system temperature interlocks or provide assurance that the shutdown cooling system is designed for full reactor temperature. By letter dated December 6, 1982, the licensee has committed to provide for testing of the temperature interlocks. The procedures will be implemented during the 1983 refueling outage.

4.26 <u>Topic VIII-2, Onsite Emergency Power Systems (Diesel Generator)</u>

10 CFR 50 (GDC 17), as implemented by SRP Sections 8.1 and 8.3.1 and Regulatory Guide 1.9, requires that onsite electric power systems shall be provided to permit functioning of components important to safety. Regulatory Guide 1.9 specifies that the standby diesel generator systems be designed so that spurious actuation of protective trips does not prevent diesel generators from performing that function.

4.26.1 Annunciators

In conjunction with a generic review of diesel generator annunciators, the staff determined that Dresden Unit 2 does not comply with current criteria as specified in IEEE Std. 279-1971. By letter dated February 2, 1979, the licensee agreed to make suitable modifications to the annunciators. These modifications were completed in 1979. No further action is required.

4.26.2 Protective Trips

The staff has determined that three diesel generator protective trips are not bypassed during accident conditions. Two of the protective trips, engine overspeed and high differential current, are acceptable for use during emergency operation so that the generator is not damaged. The other trip, underfrequency, does not meet current licensing requirements and should be bypassed during emergency operations.

A limited PRA of the importance of bypassing diesel generator trips indicates that the importance of this issue to risk is low. However, the reliability of ac power is a dominant sequence for risk at Dresden Unit 2 (based on the results of Millstone Unit 1 and Browns Ferry IREP studies). Because the importance of diesel generator availability is high, even though the improvement by bypassing these trips is small, the staff concludes that these trips should be bypassed. By letter dated September 10, 1982, the licensee indicated that modifications will be implemented to bypass the underfrequency protective trip during emergency operations for all diesel generators. The modifications will be completed during the 1983 refueling outage.

4.27 <u>Topic VIII-3.A, Station Battery Capacity Test Requirements</u>

10 CFR 50 (GDC 18), as implemented by Regulatory Guide 1.129, requires periodic testing to determine battery capacity and demonstrate that the batteries will provide sufficient power under accident conditions. The Dresden Unit 2 program for testing the batteries does not satisfy these requirements.

The limited PRA performed for this issue has determined that a loss of dc power does have an impact on the dominant sequences leading to a core-melt accident. Using the assumption that all past battery testing was ineffective, the PRA study concluded that implementation of adequate battery testing would improve battery reliability by approximately a factor of 15. Therefore, the PRA has classified this issue of high importance to risk.

The staff proposes that the testing of the batteries be in accordance with IEEE Std. 450-1975, IEEE Std. 308-1974, BTP EICSB 6, and the "Standard Technical Specifications for General Electric Boiling Water Reactors" (NUREG-0123).

By letter dated December 3, 1982, the licensee provided information regarding the existing station battery test procedures. At each refueling outage, the station batteries are subjected to a manufacturer's rated capacity discharge test to verify that the capacity is equal to or greater than 85% of the manufacturer's rating. The licensee has concluded that the station capacity discharge test as performed is more severe than the staff's recommended service test.

The staff has reviewed the information provided by the licensee and concurs with the licensee's conclusion regarding the conservatism of the existing battery test. Therefore, backfitting is not required.

4.28 Topic VIII-3.B, DC Power System Bus Voltage Monitoring and Annunciation

10 CFR 50.55a(h), through IEEE Std. 279-1971, and 10 CFR 50 (GDC 2, 4, 5, 17, 18, and 19), as implemented by SRP Section 8.3.2, Regulatory Guides 1.6, 1.32, 1.47, 1.75, 1.118, and 1.129, and BTP ICSB 21, require that the control room operator be given timely indication of the status of the batteries and their availability under accident conditions.

The Dresden Unit 2 control room does not have indication of battery voltage, battery current, battery breaker/fuse open alarm, battery charger output current, or battery charger output breaker/fuse open alarm. Therefore, the dc power system monitoring is not in compliance with current licensing criteria.

A limited PRA was performed to determine the importance to risk of dc instrumentation, indication, and alarms. It was determined that the proposed additional monitoring devices would reduce the dc bus unavailability by about a factor of 5. This reduction is due almost equally to a reduction in breaker unavailability and battery unavailability. DC power appears in some dominant accident sequences, and resolution of this issue would have a significant impact on the value of the top event in the fault tree. This issue is, therefore, of high risk importance, as discussed in Appendix D. It is the staff's position that the licensee modify the existing dc power system monitoring for breaker or fuse position and battery availability.

By letter dated October 5, 1982, the licensee provided a commitment to provide battery voltage indication in the control room. In addition, the licensee has proposed to monitor the following indications and alarms on the local printer in the control room: (1) battery current, (2) battery charger current, (3) battery breaker open alarms, and (4) battery charger (ac and dc) breaker open alarms. A dc ground alarm already exists in the control room. The staff has found the licensee's proposal acceptable.

4.29 Topic IX-5, Ventilation Systems

10 CFR 50 (GDC 4, 60, and 61), as implemented by SRP Sections 9.4.1, 9.4.2, 9.4.3, 9.4.4, and 9.4.5, requires that the ventilation systems shall have the capability to provide a safe environment for plant personnel and for engineered safety features.

4.29.1 Battery Room Ventilation

The battery room contains the batteries that provide emergency dc power essential for postaccident shutdown of the reactor. Specifically designed ventilation is considered essential to ensure removal of hydrogen generated as a result of battery charging after loss of offsite power. Following a loss-of-offsite-power event, operator action is required to reinitiate the battery room ventilation system. During that inoperative period, hydrogen is generated because of continued battery charging.

By letter dated December 13, 1982, the licensee provided information regarding the amount of hydrogen generated during battery charging. The licensee used conservative assumptions to determine the potential hydrogen accumulation, while assuming no ventilation of the area. The staff has reviewed the licensee's calculations and found that the maximum concentration of accumulated hydrogen will remain below the combustible limits. Backfitting, therefore, is not required.

4.29.2 Low-Pressure Coolant Injection (LPCI)/Core Spray and Diesel Generator Rooms

The ventilation systems for the LPCI/core spray and diesel generator rooms are subject to disabling single failures.

(1) LPCI/Core Spray Room

The LPCI and emergency core spray pumps are located in corner rooms on the basement level of the reactor building that are serviced by the reactor building ventilation system. The reactor building ventilation system can be manually supplied with emergency diesel power. In addition, each LPCI pump room contains its own room cooler. These individual units cool by means of the diesel generator cooling water system, and their fan motors are supplied by electrical motor control centers that are designated as "diesel-powered essential service." Despite provision of essential electrical service, the fans of the LPCI cubical coolers do not have the redundancy to ensure cooling in the event of a failure within the unit. However, since core cooling can be accomplished using high-pressure coolant injection (HPCI) and one core spray subsystem, the loss of ventilation in one room will not affect the ability to achieve safe shutdown. Therefore, backfitting is not recommended.

(2) Diesel Generator Rooms

DGs 2 and 2/3 are housed in separate rooms served by separate ventilation systems. Cooling is provided by the diesel service water systems, and the ventilation systems both vent the rooms and cool associated switchgear equipment. Each DG room is ventilated by a single 30-hp fan that is automatically loaded to an essential service motor control center powered by its respective DG. Outside air and/or turbine building air is supplied to the fan through a set of temperature-controlled dampers. If either ventilation fan were to fail and result in the failure of its respective DG, the other DG would be sufficient to supply all necessary safety-related loads. In addition, the large double doors between the turbine building and DG 2 could be opened to promote natural convection. Access to the DG 2/3 building is through two single inseries doors, both of which would have to be opened to provide natural convection.

The limited PRA evaluation performed for this issue was based on the IREP study of Millstone Unit 1. During that review, no potential system failures resulting from support system ventilation failures were identified. On the basis of a review of the Dresden Unit 2 plant configuration, it was determined that the Millstone Unit 1 results are applicable to Dresden Unit 2. Therefore, the PRA has classified this issue's importance to risk as low. Backfitting, therefore, is not required.

4.30 Topic XV-1, Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve

10 CFR 50.34 requires that each applicant for a construction permit or operating license provide an analysis and evaluation of the design and performance of

structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility, including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility.

10 CFR 50 (GDC 10 and 15), as implemented by SRP Sections 15.1.1 through 15.1.4, requires that plants be adequately designed to mitigate the consequences of feedwater system malfunctions that result in an increase in feedwater flow.

The staff review of a feedwater controller failure has determined that the acceptance criteria are met only if the turbine bypass system is operable. Currently, the licensee does not have Technical Specifications that require surveillance of the turbine bypass system or that limit the reactor power or minimum critical power ratio (MCPR) when the turbine bypass system is found to be inoperable. Because the feedwater controller failure with failure of the turbine bypass may be a limiting transient, exceeding the fuel design limits could result. It is also possible that another transient limits MPCR or reactor power and no change is required.

The staff concludes that analysis of feedwater controller failure without bypass should not be required for the current fuel cycle for the following reasons:

(1) The licensee currently plans to shut down in early 1983 for refueling. The licensee will perform a reload analysis for the new core configuration before startup. This analysis will include an evaluation of anticipated transients. If credit is taken in the reload analysis for operability of turbine bypass, the staff will require appropriate surveillance of the turbine bypass valves and limits for reactor power or MCPR if the turbine bypass is found inoperable. Technical Specifications will be developed and reviewed as part of the core reload evaluation to reflect the fuel vendor and cycle-specific characteristics of the core.

(2) PRA studies of BWRs indicate that feedwater controller transients without bypass are of low importance to risk.

Backfitting, therefore, is not recommended.

4.31 Topic XV-16, Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment

10 CFR 100, as implemented by SRP Section 15.6.2, requires that the radiological consequences of failure of small lines carrying primary coolant outside containment be limited to small fractions of the exposure guidelines of 10 CFR 100.

The staff has determined that Dresden Unit 2 does not comply with current acceptance criteria. Based on the existing Technical Specification limits for primary coolant activity, the potential offsite doses would substantially exceed the applicable dose limits. It is the staff's position that reactor coolant activity limits should be maintained within the limits imposed on new operating reactors, that is, within the limits of the Standard Technical Specifications (STS) for General Electric Boiling Water Reactors (NUREG-0123). This is necessary to limit plant operation with potentially significant amounts of failed fuel so that the radiological consequences of events that do not further damage fuel but do involve a release of reactor coolant to the environment will be low.

Reducing reactor coolant activity to the STS level would not result in calculated doses within the limits specified in current licensing criteria; however, the doses are within the guidelines of 10 CFR 100. Therefore, since the offsite dose consequences are within the guidelines of 10 CFR 100 and the probability of failing the line before the isolation valve and excess flow check valve is low, it is the staff's position that backfitting the BWR STS limits for reactor coolant activity is sufficient to ensure that the radiological consequences to the environment from a failure of small lines are acceptably low.

The limited PRA for Dresden Unit 2 has classified this issue's importance to risk as low. This is due to the overwhelming portion of risk from core-melt accidents. However, because of the significant radiological impact resulting from this accident in the absence of core melt, it is the staff's position that primary coolant activity be maintained within acceptable limits.

By letter dated December 6, 1982, the licensee has committed to propose plant Technical Specifications that will limit the primary coolant iodine activity to levels corresponding to the STS level. The specifications will include action statements and surveillance requirements based on plant-specific operation requirements. The staff finds this proposal acceptable.

4.32 Topic XV-18, Radiological Consequences of a Main Steam Line Failure Outside Containment

10 CFR 100, as implemented by SRP Section 15.6.4, requires that the radiological consequences of failure of a main steam line outside containment be limited to small fractions of the exposure guidelines of 10 CFR 100. On the basis of an independent assessment of the radiological consequences of a main steam line failure outside containment, the staff has determined that Dresden Unit 2 does not meet the current acceptance criteria. If the existing Technical Specification limits for primary coolant activity are used, the potential offsite doses would exceed the applicable dose limits. It is the staff's position that the licensee should maintain the primary coolant activity within the General Electric STS limits, which would meet the acceptance criteria. Since the staff's analysis shows that the small-line failure is more limiting than the main steam line failure, resolution of Topic XV-16 will also resolve the concerns of Topic XV-18.

Table 4.1 Integrated assessment summary

SEP Topic No.	Section No.	Title	Tech. Spec. modifications required from SEP review	Backfit requirements	Licensee agrees	Completion date	PRA rating
II-3.B, II-3.B.1	4.1.1	Design-Basis Groundwater Level	No	None	-	-	
II-3.C	4.1.2	Probable Maximum Flood	No	See Section 4.1.4.	-	-	-
	4.1.3	Roof Loadings	No	Modify parapets to ensure ponded water is within structural capacity of roof.	Yes	To be provided	-
	4.1.4	Flood Emergency Plan	No	Modify existing procedures to address ability to cope with probable maximum flood.	Yes	1983 refueling outage	-
III-1	4.2.1	Radiography Requirements	No	(1) Identify Class 2 vessels built to Class C requirements containing Class C joints and their examination techniques.	Yes	To be provided	-
				(2) Describe examination given to recirculation pump casing.	Yes	To be provided	
	4.2.2	Fracture Toughness	No	Demonstrate fracture toughness for components or demonstrate failure is acceptable.	Yes	To be provided	-
III-2	4.3.1	Reactor Building Structure Above the Operating Floor	No	None	-	-	-
	4.3.2	Ventilation Stack	No	Ensure stack failure will not affect safe shutdown	Yes	To be provided	-

Table 4.1 (Continued)

SEP Topic No.	Section No.	Title	Tech. Spec. modifications required from SEP review	Backfit requirements	Licensee agrees	Completion date	PRA rating
III-2	4.3.3	Components Not Enclosed in Qualified Structures	No	Identify and ensure components can withstand tornado loading or their loss will not affect safe operation.	-	-	-
	4.3.4	Roof Decks	No	Demonstrate failure of roof decks will not affect plant safety.	Yes	11/22/82	-
	4.3.5	Load Combinations	No	Will be addressed in Topic III-7.B.	-	-	-
III-3.C	4.4.1	Flow-Regulation Station	No	None	-	-	-
	4.4.2	Intake and Discharge Structures	No	None	-	-	-
	4.4.3	Inspection Program	No	Modify procedures to ensure (1) supervision by qualified personnel and (2) inspecions following extreme events.	Yes	1983 refueling outage	-
III-4.A	4.5.1	Service Water System (SWS)	No	Demonstrate auxiliary electrical equipment room has adequate ventilation as part of TMI control room habitability.	Yes	Part of TMI Action Plan	-
	4.5.2	Station Battery Systems	No	None	-	-	-
	4.5.3	Diesel Generator Ventilation	No	Ensure that DG 2 and DG 2/3 will remain operable if ventilation is lost.	Yes	12/82	-

Table 4.1 (Continued)

SEP Topic No.	Section No.	Title	Tech. Spec. modifications required from SEP review	Backfit requirements	Licensee agrees	Comple-tion date	PRA rating
III-4.A	4.5.4	Exterior Tanks	No	Ensure safe shutdown can be accomplished using missile-protected systems.	Yes	12/82	-
III-4.B	4.6	Turbine Missiles	No	Provide schedule and basis for inspection of low-pressure turbines.	Yes	10/8/82	-
III-5.A	4.7.1	Jet Impingement on Target Pipe	No	Evaluate and address effects of jet impingement regardless of ratio of pipe sizes.	Yes	11/17/82	-
	4.7.2	Broken-Pipe Impact on Target Pipe	No	Demonstrate deformation associated with global strain would not affect functionability of target pipe.	Yes	11/17/82	-
	4.7.3	Detectability Requirements	No	Ensure detectability for through-wall cracks of high-energy piping systems.	Yes	11/17/82	-
	4.7.4	Criteria Implementation	No	(1) Provide criteria and results for pipe whip load formulation.	Yes	11/17/82	-
				(2) Ensure pipe whip and jet impingement will not affect containment liner.	Yes	11/17/82	-
III-5.B	4.8	Pipe Break Outside Containment	No	None	-	-	Low

Table 4.1 (Continued)

SEP Topic No.	Section No.	Title	Tech. Spec. modifications required from SEP review	Backfit requirements	Licensee agrees	Completion date	PRA rating
III-6	4.9.1	Piping Systems	No	Addressed as part of IE Bulletin 79-14 effort.	-	-	-
	4.9.2	Mechanical Equipment	No	(1) Supply information regarding motor-operated-valve (MOV) lever arms and limiting moments.	Yes	11/3/82	-
				(2) Staff will use Oyster Creek analysis to determine seismic capability of reactor internal supports.	Yes	After Oyster Creek submittal	-
				(3) Provide further information regarding seismic capability of recirculation pump and supports.	Yes	After staff-review of pipe break	
	4.9.3	Qualification of Cable Trays	No	To be determined following completion of Owners Group program.	Yes	To be provided	-
	4.9.4	Ability of Safety-Related Equipment To Function	No	SEP Owners Group program will develop USI A-46 criteria.	Yes	To be provided	-
III-7.B	4.10	Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria	No	Staff is currently reviewing licensee's submittal.	-	-	-
III-8.A	4.11	Loose-Parts Monitoring and Core Barrel Vibration Monitoring	No	None	-	-	Low

Table 4.1 (Continued)

SEP Topic No.	Section No.	Title	Tech. Spec. modifications required from SEP review	Backfit requirements	Licensee agrees	Completion date	PRA rating
III-10.A	4.12.1	Thermal Overloads	No	Bypass thermal overload protection of MOV or ensure setpoint adequacy.	Yes	To be provided	Medium
	4.12.2	Torque Switches	No	None	-	-	-
V-5	4.13.1	System Sensitivity	No	Evaluate leakage detection capability in conjunction with pipe break inside containment.	Yes	After staff review of pipe break	Low
	4.13.2	Seismic Qualification	No	Demonstrate reliability and provide procedures corresponding to seismic events.	Yes	After staff review of pipe break	Low
	4.13.3	System Testability	No	None	-	-	Low
V-6	4.14	Reactor Vessel Integrity	No	None	-	-	-
V-10.B	4.15	Residual Heat Removal System Reliability	No	See Topic VII-3.	-	-	-
V-11.A	4.16	Requirements for Isolation of High- and Low-Pressure Systems	No	None	-	-	Low
V-11.B	4.17	Residual Heat Removal System Interlock Requirements	No	See Topic VII-3.	-	-	-

Table 4.1 (Continued)

SEP Topic No.	Section No.	Title	Tech. Spec. modifications required from SEP review	Backfit requirements	Licensee agrees	Completion date	PRA rating
VI-4	4.18.1	Locked-Closed Valves	No	Provide mechanical locking devices and administrative procedures to ensure valve closure.	Yes	1983 refueling outage	Low
	4.18.2	Leakage Detection	No	Modify procedures for postaccident leakage.	Yes	To be provided	Low
	4.18.3	Manual Isolation Valves	No	Provide locking devices for valves.	Yes	To be provided	Low
	4.18.4	Check Valves as Isolation Valves	No	None	-	-	Low
	4.18.5	Valve Location	No	None	-	-	Low
	4.18.6	Branch Lines With Single Isolation Valves	No	Provide second locked-closed valve.	Yes	To be provided	Low
VI-6	4.19	Containment Leak Testing	No	None	-	-	Low
VI-7.A.4	4.20	Core Spray Nozzle Effectiveness	No	Being reviewed as matter related to Generic Issue A-16.	-	-	-
VI-7.C.1	4.21.1	Breaker Adequacy	No	Verify adequacy of protective relaying.	Yes	To be provided	Low[1]

[1]PRA has assumed existence of appropriate procedures. Staff has found procedures inadequate.

Table 4.1 (Continued)

SEP Topic No.	Section No.	Title	Tech. Spec. modifications required from SEP review	Backfit requirements	Licensee agrees	Comple-tion date	PRA rating
VI-7.C.1	4.21.2	Disconnect Links	No	Provide procedures to verify disconnect links are open.	Yes	1983 refueling outage	Low[1]
	4.21.3	Use of Breakers During Power Operations	No	Provide administrative control to ensure breakers are not used during power operations.	Yes	1983 refueling outage	Low[1]
	4.21.4	Operation With Failed Battery	Yes	Limit time for operation with failed battery.	Yes	To be provided	-
	4.21.5	Isolation of Class 1E Sources From Non-Class 1E Loads	No	Provide short-circuit analysis to demonstrate adequate isolation.	Yes	To be provided	-
VI-10.A	4.22	Testing of Reactor Trip System and Engineered Safety Features, Includ-ing Response-Time Testing	No	None	-	-	Low
VI-10.B	4.23.1	Sharing of DC Systems	No	Prohibit paralleling of shared dc systems during power operations.	Yes	To be provided	Low
	4.23.2	Diesel Generator Bypass	No	Prohibit placing DG 2/3 switch in "bypass" during normal operations.	Yes	Complete	-
	4.23.3	Battery Status Indication	-	See Topic VIII-3.B	-	-	-
	4.23.4	Battery Room Ventilation	-	See Topic IX-5	-	-	-

[1]PRA has assumed existence of appropriate procedures. Staff has found procedures inadequate.

Table 4.1 (Continued)

SEP Topic No.	Section No.	Title	Tech. Spec. modifications required from SEP review	Backfit requirements	Licensee agrees	Completion date	PRA rating
VII-1.A	4.24.1	Reactor Protection System (RPS) Control Systems	No	Ensure common-mode electrical faults will not disable neutron flux monitoring systems.	Yes	To be provided	Low
	4.24.2	Process Computer	No	Ensure RPS is protected from common-mode electrical faults.	Yes	To be provided	-
	4.24.3	RPS Channel Power Supplies	No	Install Class 1E protection at interface of RPS and RPS power supply.	Yes	1983 refueling outage	-
VII-3	4.25.1	Procedures for Shutdown From Outside Control Room	No	Provide procedures for achieving cold shutdown from outside control room.	Yes	As part of fire protection	-
	4.25.2	Use of Safety-Grade Systems	No	Revise procedures to achieve cold shutdown using safety-grade systems.	Yes	Complete	-
	4.25.3	Residual Heat Removal Single-Failure Criteria	No	None	-	-	-
	4.25.4	Inservice Testability	No	Provide procedures for testing of shutdown cooling system temperature interlocks.	Yes	1983 refueling outage	Medium
VIII-2	4.26.1	Annunciators	No	Modify annunciators to comply with IEEE Std. 279-1971.	Yes	Complete	-

Table 4.1 (Continued)

SEP Topic No.	Section No.	Title	Tech. Spec. modifications required from SEP review	Backfit requirements	Licensee agrees	Completion date	PRA rating
VIII-2	4.26.2	Protective Trips	No	Bypass DG underfrequency trip.	Yes	1983 refueling outage	Low
VIII-3.A	4.27	Station Battery Capacity Test Requirements	No	None	-	-	High
VIII-3.B	4.28	DC Power System Bus Voltage Monitoring and Annunciation	No	Modify existing dc power system monitoring for breaker or fuse position and battery availability.	Yes	To be provided	High
IX-5	4.29.1	Battery Room Ventilation	No	None	-	-	-
	4.29.2	Low-Pressure Coolant Injection (LPCI)/Core Spray and Diesel Generator Rooms	No	None	-	-	Low
XV-1	4.30	Increase in Feedwater Flow	No	None	-	-	Low
XV-16	4.31	Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment	Yes	Modify plant Technical Specifications limits on primary coolant iodine activity.	Yes	To be provided	Low
XV-18	4.32	Radiological Consequences of a Main Steam Line Failure Outside Containment	Yes	Modify plant Technical Specification limits on primary coolant iodine activity.	Yes	To be provided	Low

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---, Dec. 11, 1980, from R. F. Janacek (CECo) to T. A. Ippolito (NRC), Subject: Dresden Station Units 2 and 3, Quad Cities Station Units 1 and 2, RPS Power Supply Modifications - NRC Docket Nos. 50-237/249 and 50-254/265.

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---, Mar. 9, 1982, from P. O'Connor, (NRC) to L. DelGeorge (CECo), Subject: SEP Topic III-1, Quality Group Classification of Components and Systems (Dresden Unit 2).

---, Mar. 31, 1982, from T. J. Rausch (CECo) to H. R. Denton (NRC), Subject: Dresden Station Units 2 and 3, Quad Cities Station Units 1 and 2, Proposed Amendment to Technical Specifications.

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---, July 7, 1982, from T. J. Rausch (CECo) to P. O'Connor (NRC), Subject: Dresden 2 - SEP Topic III-6, Seismic Design Considerations.

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---, Aug. 31, 1982, from P. B. Fieldler (Jersey Central Power & Light Company) to D. M. Crutchfield (NRC), Subject: Oyster Creek Nuclear Generating Station License DPR-16 - Docket No. 50-219 - SEP Topic III-5.A, "Effects of Pipe Breaks on Structures, Systems, and Components Inside Containment."

---, Sept. 2, 1982(a), from P. W. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic III-1, Quality Group Classification of Components and Systems - Dresden Nuclear Power Station Unit 2.

---, Sept. 2, 1982(b), from T. J. Rausch (CECo) to P. O'Connor (NRC), Subject: Dresden 2 - SEP Topic: III-3.C, "Inservice Inspection of Water Control Structures."

---, Sept. 3, 1982, from T. J. Rausch (CECo) to P. O'Connor (NRC), Subject: Dresden 2 - SEP Topic III-6, Seismic Design Considerations.

---, Sept. 10, 1982, from T. J. Rausch (CECo) to P. O'Connor (NRC), Subject: Dresden 2 - SEP Topic VIII-2, Onsite Emergency Power Systems - Diesel Generator.

---, Sept. 15, 1982, from T. J. Rausch (CECo) to P. O'Connor (NRC), Subject: Dresden 2 - SEP Topic: V-11.A, Requirements for Isolation of High and Low Pressure Systems.

---, Sept. 24, 1982, from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic VI-4, Containment Isolation System Dresden Nuclear Power Station, Unit 2.

---, Oct. 5, 1982, from T. J. Rausch (CECo) to P. O'Connor (NRC), Subject: Dresden 2 SEP Topic: VIII-3.B, DC Power System Bus Voltage Monitoring and Annunciation.

--- Oct. 8, 1982, from T. J. Rausch (CECo) to P. O'Connor (NRC), Subject: Dresden 2 SEP Topic: III-4.B, Turbine Missiles.

---, Oct. 19, 1982, from T. J. Rausch (CECo) to P. O'Connor (NRC), Subject: Dresden 2 SEP Topic: V-11.A, Requirements for Isolation of High and Low Pressure Systems.

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---, NUREG-0800 (formerly NUREG-75/087), "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1981 (includes Branch Technical Positions).

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---, Regulatory Guide (RG) 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems."

---, RG 1.9, "Selection, Design and Qualification of Diesel-Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants."

---, RG 1.11, "Instrument Lines Penetrating Primary Reactor Containment."

---, RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste Containing Components of Nuclear Power Plants."

---, RG 1.27," Ultimate Heat Sink for Nuclear Power Plants."

---, RG 1.32, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants."

--, RG 1.41, "Preoperational Testing of Redundancy Onsite Electric Power Systems To Verify Proper Load Group Assignments."

---, RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems."

---, RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems."

---, RG 1.59, "Design Basis Floods for Nuclear Power Plants."

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---, RG 1.81, "Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants."

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APPENDIX A

TOPIC DEFINITIONS FOR SEP REVIEW*

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*The topic definitions and other data appearing in this appendix were assembled in April 1977; therefore, some references to organizations and other references reflect the status of the review at that time. The basis for deletion of a topic because the review of a related TMI task, USI, or other SEP topic was identical to the review of the SEP topic was developed in May 1981 on a generic basis and does not address the plant-specific design aspects. The plant-specific deletions that are due to generic review or nonapplicability to the Dresden Unit 2 design are given in Appendices B and C.

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TOPIC: II-1.A Exclusion Area Authority and Control

(1) Definition:

The establishment of the exclusion area and the licensee's control over it are reviewed at the construction permit/operating license stage. Thereafter, the licensees are required to report any changes with safety implications. The concern exists, however, that (1) the original review may not have been as thorough as currently done, or (2) changes may have occurred but have not been reported and reviewed. In particular, new activities within the exclusion area (for example, new recreational facilities or offshore oil drilling) and topographical changes (for example, changes in water levels) may need to be reviewed.

(2) Safety Objective:

To assure that appropriate exclusion area authority and control is maintained by the licensee.

(3) Status:

Selective reviews have been performed (San Onofre Nuclear Generating Station Unit 1) or are under way (Fort Calhoun) where changes in exclusion area boundary have become necessary.

(4) References:

1. Title 10, "Energy," Code of Federal Regulations, Part 100*
2. NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition, "December 1975,"** Section 2.1.2

TOPIC: II-1.B Population Distribution

(1) Definition:

Population distribution in the vicinity of operating plants may have changed since the initial review was performed at the construction permit stage. Special attention should be given to new housing and commercial, military, or institutional installations established since the initial population-distribution review.

(2) Safety Objective:

New population distributions may require revision of low-population zone (LPZ) and population center to assure appropriate protection for the public by complying with the guidelines of 10 CFR Part 100. Adjustments may have

*Hereafter referred to as 10 CFR.
**Hereafter referred to as Standard Review Plan.

to be made in emergency plans. New accident analyses may have to be performed to determine consequent conformance with 10 CFR Part 100 at new LPZ distances. Potential need for additional engineered safety features (for example, chemical sprays or better filters) exists.

(3) Status:

Has been done on a selective basis only, that is, Pilgrim Unit 1 new population center.

(4) References:

1. 10 CFR Part 100
2. Standard Review Plan, Section 2.1.3

TOPIC: II-1.C Potential Hazards or Changes in Potential Hazards Due to Transportation, Institutional, Industrial, and Military Facilities

(1) Definition:

For operating plants there are three concerns:

(a) New hazards created since the facility was licensed,

(b) Hazards considered for licensing but that have expanded beyond projections or which were not reviewed against current criteria, and

(c) Hazards that were not analyzed at the licensing stage because of lack of regulatory criteria at the time.

Nearby transportation, institutional, industrial, and military facilities may be threats to safe plant operation due to:

(a) Control room infiltration of toxic gases,

(b) Onsite fires triggered by transport of combustible chemicals from offsite releases,

(c) Shock waves due to detonation of stored or transported explosives and military ordnance firing, and

(d) Onsite aircraft impact.

(2) Safety Objective:

To assure that the control room is habitable at all times and that the postulated hazards will not result in releases in excess of the 10 CFR Part 100 guidelines by disabling systems required for safe plant shutdown.

(3) Status:

Action has been taken on a selective basis only, for example, curbing of military air activity in the vicinity of the Big Rock Point Plant. Liquid natural gas (LNG) hazards at Calvert Cliffs are under review. The review of older plants did not consider offsite hazards in detail (for example, aircraft traffic in the vicinity).

(4) Reference:

Standard Review Plan, Sections 2.2.1 and 2.2.2

TOPIC: II-2.A Severe Weather Phenomena

(1) Definition:

Safety-related structures, systems, and components should be designed to function under all severe weather conditions to which they may be exposed. Meteorological phenomena to be considered include tornadoes, snow and ice loads, extreme maximum and minimum temperatures, lightning, combinations of meteorology and air-quality conditions contributing to high corrosion rates, and effects of sand and dust storms.

(2) Safety Objective:

To assure that the designs of safety-related structures, systems, and components reflect consideration of appropriate extreme meteorological conditions and severe weather phenomena. This effort would identify deficiencies in designs and/or operation that may contribute to accidental releases of radioactivity to the atmosphere resulting in doses to the public in excess of 10 CFR Part 100 or Part 20 guidelines (as appropriate to the design of the component or system).

(3) Status:

Generic studies have been initiated to develop guidelines for extreme temperatures and lightning, and to the review the current Branch Positions on snow loads. Estimated completion dates are 6/1/78 or later.

(4) References:

1. 10 CFR Part 100 or Part 20
2. Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants"
3. Standard Review Plan, Section 2.3.1
4. Branch Technical Position, "Winter Precipitation Loads," March 24, 1975
5. Inquiry by Chairman Rowden Concerning Lightning Protection, July 9, 1976
6. 10 CFR Part 50

TOPIC: II-2.B Onsite Meteorological Measurements Program

(1) Definition:

To review the onsite meteorological measurements program to determine the extent that the licensee complies with 10 CFR Part 50, Appendix E and Appendix I.

(2) Safety Objective:

To assure that adequate meteorological instrumentation to quantify the offsite exposures from routine releases is available and maintained.

(3) Status:

Onsite meteorological measurements programs are being reviewed as a part of the Appendix I evaluations.

(4) References:

1. 10 CFR Part 50, Appendix E and Appendix I
2. Regulatory Guide 1.97, Rev. 1, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident"
3. Regulatory Guide 1.23, "Onsite Meteorological Programs"
4. Standard Review Plan, Section 2.3.3

(5) Basis for Deletion (Related TMI Task, Unresolved Safety Issue (USI), or Other SEP Topic):

(a) TMI Action Plan Task II.F.3, "Instrumentation for Monitoring Accident Conditions" (NUREG-0660)

Task II.F.3 requires that appropriate instrumentation be provided for accident monitoring with expanded ranges and a source term that considers a damaged core capable of surviving the accident environment in which it is located for the length of time its function is required. Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident," issued December 1980, contains the required meteorological instrumentation to quantify the offsite exposure.

(b) TMI Action Plan Task III.A.1, "Improve Licensee Emergency Preparedness - Short Term" (NUREG-0660)

Task III.A.1 requires the evaluation of 10 CFR Part 50, Appendix E, backfit requirements in accordance with NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." Backfit requirements include review of the Onsite Meteorological Measurement Program.

The evaluations required by Tasks II.F.3 and III.A.1 are identical to SEP Topic II-2.B; therefore, this SEP topic has been deleted.

TOPIC: II-2.C Atmospheric Transport and Diffusion Characteristics for Accident Analysis

(1) Definition:

To review the atmospheric transport and diffusion characteristics assumed to demonstrate compliance with the 10 CFR 100 guidelines with respect to plant design, control room habitability, and doses to the public during and following a postulated design-basis accident. This effort would examine the assumptions for:

(a) Effects of explosive concentrations from onsite or offsite releases of hazardous material for consideration in structural design,

(b) Calculation of relative concentration (x/Q) values for releases of radioactivity and toxic chemicals for consideration in control room habitability, and

(c) Calculations of doses to the public resulting from releases of radioactivity to the atmosphere during and following a postulated design-basis accident.

This effort is considered necessary because most original reviews were performed using the assumptions provided in Regulatory Guides 1.3 and 1.4 which have been found to be generally nonconservative based on evaluation of over 50 sites with actual meteorological observations.

(2) Safety Objective:

To assure that the atmospheric transport and diffusion characteristics originally assumed to demonstrate compliance with the 10 CFR 100 guidelines are appropriate, considering additional onsite meteorological data and results of recent atmospheric diffusion experiments.

(3) Status:

A review of long-term (annual average) atmospheric transport and diffusion characteristics is ongoing for Appendix I evaluations independent of the SEP effort. A study has also recently been performed by the Hydrology-Meteorology Branch for the Division of Operating Reactors for review of the meteorological assumptions for estimating control room dose consequences resulting from post-LOCA purges through tall stacks.

(4) References:

1. 10 CFR Part 20
2. 10 CFR Part 50, Appendix A and Appendix I
3. 10 CFR Part 100
4. Regulatory Guides
   1.3, "Assumption Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors"
   1.4, "Assumptions Used for Evaluating the Potential Radiologcal Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors"
5. Standard Review Plan, Sections 2.3.4, 6.4, 2.2.1, 2.2.2, and 2.2.3

TOPIC: II-2.D Availability of Meteorological Data in the Control Room

(1) Definition:

Data from the onsite meteorological program should be available in the control room.

(2) Safety Objective:

To assure that the lincensee has appropriate meteorological logical data displayed in the control room to assess conditions during and following an accident to allow for (1) early indication of the need to initiate action necessary to protect portions of the offsite public and (2) an estimate of the magnitude of the hazard from potential or actual accidental releases.

(3) Status:

No work currently being done on this subject for operating plants.

(4) References:

1. 10 CFR Part 50, Appendix E and Appendix I
2. Regulatory Guide 1.97, Rev. 1, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident"
3. Regulatory Guide 1.23, "Onsite Meteorological Programs"
4. Standard Review Plan, Section 2.3.3

(5) Basis for Deletion (Related TMI Task, USI, or Other SEP Topic):

(a) TMI Action Plan Task II.F.3, "Instrumentation for Monitoring Accident Conditions" (NUREG-0660)

Task II.F.3 requires that appropriate instrumentation be provided for accident monitoring with expanded ranges and a source term that considers a damaged core capable of surviving the accident environment in which it is located for the length of time its function is required. Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident," issued December 1980, contains the required meteorological instrumentation to quantify the offsite exposure.

(b) TMI Action Plan Task III.A.1, "Improve Licensee Emergency Preparedness - Short Term" (NUREG-0660)

Task III.A.1, "Improve Licensee Emergency Preparedness - Short Term," requires the evaluation of 10 CFR Part 50, Appendix E backfit requirements in accordance with NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." Backfit requirements include review of the Onsite Meteorological Measurement Program.

(c) TMI Action Plan Task I.D.1, "Control Room Design Reviews" (NUREG-0660)

Task I.D.1, "Control Room Design Reviews," requires that operating reactor licensees and applicants for operating licenses perform a detailed control room design review to identify and correct design deficiencies. This review will include an assessment of control room layout, the adequacy of the information provided, the arrangement and identification of important controls and instrumentation displays, the usefulness of the audio and visual alarm systems, the information recording and recall capability, lighting, and other considerations of human factors that have an impact on operator effectiveness.

The evaluations required by Tasks II.F.3, III.A.1, and I.D.1 are indentical to SEP Topic II-2.D; therefore, this SEP topic has been deleted.

TOPIC: II-3.A Hydrologic Description

(1) Definition:

Hydrologic considerations are the interface of the plant with the hydrosphere, the identification of hydrologic causal mechanisms that may require special plant design or operating limitations with regard to floods and water supply requirements, and the identification of surface- and groundwater uses that may be affected by plant operation.

These hydrologic considerations may have changed since they were reviewed at the licensing stage. A review of such changes, if any, should be performed including an assessment of their impact on the plants.

(2) Safety Objective:

To assure that the designs of safety-related structures, systems, and components reflect consideration of appropriate hydrologic conditions, and to identify deficiencies in designs and/or operations that could contribute to accidental radioactive releases.

(3) Status:

No work currently being done on this subject for operating plants.

(4) References:

1. 10 CFR Parts 20, 50, and 100
2. American National Standards Institute, ANSI N170-1976, "Standards for Determining Design Basis Flooding at Power Reactor Sites"
3. Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants"
4. Standard Review Plan, Section 2.4.1

TOPIC: II-3.B Flooding Potential and Protection Requirements

(1) Definition:

If the potential for floods exists and protection is required, the type of protection (sand bags, flood doors, bulkheads, and so forth) will be reviewed to assure that equipment is available and that provisions have been made to implement the required protection.

(2) Safety Objective:

To assure that safety-related structures, systems, and components are adequately protected against floods.

(3) Status:

Flooding protection requirements were reviewed on selected operating plants during the winter of 1976 due to the potential for flooding caused by ice accumulation and predictions for abnormally high spring runoff for some areas.

(4) References:

1. 10 CFR Parts 50 and 100
2. Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants"
3. American National Standards Institute, ANSI N170-1976, "Standards for Determining Design Basis Flooding at Power Reactor Sites"
4. Standard Review Plan, Section 2.4.10

TOPIC: II-3.B.1 Capability of Operating Plants To Cope With Design-Basis Flooding Conditions

(1) Definition:

Protection against postulated floods is accomplished, if necessary, by "hardening" the plant and by implementing appropriate technical specifications and emergency procedures.

These technical specifications and flood emergency procedures need to be reviewed for plants licensed prior to 1972 to establish the degree of conformance with current criteria. Flooding criteria used for the design of older plants are not known.

(2) Safety Objective:

Same as II-3.B

(3) Status:

Same as II-3.B

(4) References:

1. 10 CFR Part 100
2. American National Standards Institute, ANSI N170-1976, "Standards for Determining Design Basis Flooding at Power Reactor Sites"
3. Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants"
4. Standard Review Plan, Sections 2.4.3, 2.4.4, 2.4.5, and 2.4.7

TOPIC: II-3.C Safety-Related Water Supply (Ultimate Heat Sink [UHS])

(1) Definition:

To determine the adequacy of onsite water sources with respect to providing safety-related water during emergency shutdown and maintenance of safe shutdown. The location and inventory of safety-related water sources and the meteorological conditions to be used in evaluating both temperature and inventory of the sources should be established. Considerations of ice, low water, leak potential, and underwater dams should be included. In most cases, plants operating prior to 1973 will have to be reviewed to establish the degree of conformance with current criteria. Prior to the issuance of Regulatory Guide 1.27 in 1973, the Standard Format and Content (now Regulatory Guide 1.70) provided the only guidelines to prospective applicants on UHS requirements. Since compliance was not required and hydrologic and meteorologic criteria had not been established, usually only minimal data were provided.

(2) Safety Objective:

To assure an appropriate supply of cooling water during normal and emergency shutdown procedures.

(3) Status:

No work currently being done on this subject for operating plants.

(4) References:

1. 10 CFR Part 100
2. Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants"
3. Standard Review Plan, Sections 2.4.11 and 9.2.5

TOPIC: II-4 Geology and Seismology

(1) Definition:

Prior to the adoption of Appendix A to 10 CFR Part 100 in 1973, the Standard Format provided the only guidelines to prospective applicants regarding the type of geologic and seismic information needed by the Atomic Energy Commission staff. The applicant, because compliance with Regulatory Guide 1.70 was not required, usually provided only minimal data. Therefore, a re-review of plants licensed prior to 1973 is needed in order to determine the adequacy of the plant design with respect to geologic and seismologic phenomena such as earthquakes, landslides, ground collapse, and liquefaction.

The review will also include ground motion and surface faulting and will establish the ground-motion values and foundation conditions to be input into the structural reevaluation for seismic loads. (It is possible that some of the older plants would require assessing only the effects of new geologic and seismic discoveries on the site safety and the resulting design acceleration and/or the response spectra.)

(2) Safety Objective:

To assure that accidents (for example, loss-of-coolant accident) do not occur and that plants can safely shut down in the event of geologic and seismologic phenomena which may occur at the site.

(3) Status:

Selected plants are undergoing reevaluation of geology and seismology (San Onofre Nuclear Generating Station Unit 1 and Humboldt Bay). A plan for reevaluating operating plants was developed in 1975-76 but has not been implemented pending formation of the Systematic Evaluation Program.

(4) References:

1. Standard Review Plan, Sections 2.5.1, 2.5.2, 2.5.3, 2.5.4, and 2.5.5
2. 10 CFR Part 100, Appendix A

TOPIC: II-4.A Tectonic Province

(1) Definition:

This subtopic covers a specific area within the major topic Geology and Seismology. Its purpose is to reassess the tectonic province for operating plants based on more current knowledge. (A tectonic province is a region characterized by a relative consistency of the geologic structural features contained within. Tectonic provinces are used operationally as regions within which risk from earthquakes not associated with tectonic structures or faults is considered uniform. Usually the largest historical earthquake not associated with a specific structure can be assumed to occur anywhere within the same province.)

(2) Safety Objective:

To assure that plants can be safely shut down in the event of geologic and seismologic phenomena which may occur at the site.

(3) Status:

The Geosciences Branch is currently attempting to delineate the boundaries of specific tectonic provinces (estimated completion date, fall 1977). The Site Safety Standards Branch is attempting to revise Appendix A to 10 CFR Part 100 so that the definition of tectonic province will more closely conform to its operational use (estimated completion date, 1978). We currently accept such provinces as generally proposed by King, Rogers, or Eardley. Limited subdivision of these provinces has been allowed based on thorough geological and seismic analyses.

(4) References:

1. 10 CFR Part 100, Appendix A
2. King, P. B., Tectonic Map of North America; Washington, D.C., U.S. Geological Survey, 1969
3. Rogers, John, The Tectonics of the Appalachians, N.Y., Wiley-Interscience, 271 p, 1970
4. Eardley, A. H., "Tectonic Divisions of North America," Bulletin of the American Association of Petroleum Geologists, 35: 2229-2237, 1951

TOPIC: II-4.B Proximity of Capable Tectonic Structures in Plant Vicinity

(1) Definition:

This subtopic covers a specific area within the major topic Geology and Seismology. Its purpose is to determine the expected shaking characteristics at a plant site from known capable faults. The ground motion associated with an earthquake generated by a capable fault or a tectonic structure may be greater than that associated with earthquakes in the same tectonic province not related to the structure.

(2) Safety Objectives:

To assure that plants can be safely shut down in the event of geologic and seismologic phenomena which may occur at the site.

(3) Status:

No work currently being done on this subject for operating plants.

(4) References:

1. 10 CFR Part 100, Appendix A
2. Standard Review Plan, Section 2.5.2
3. Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants"

TOPIC: II-4.C Historical Seismicity Within 200 Miles of Plant

(1) Definition:

Determination of the safe shutdown earthquake (SSE) is made with consideration of past seismicity in the vicinity of the plant. However, there is sometimes disagreement or inconsistency in reporting older earthquakes in the literature. Current high seismicity may also indicate possible hidden tectonic features.

The historical seismicity within 200 miles of the plants will be reviewed including all earthquakes of Richter magnitude greater than 3.0 or of Modified Mercalli intensity greater than III. Association with tectonic features and provinces should be included.

(2) Safety Objective:

To assure that the SSE is compatible with past seismicity in the area.

(3) Status:

No work currently being done in this subject for operating reactors.

(4) References:

1. Richter, C. F., Elementary Seismology, W. H. Freeman and Company, San Francisco, Calif., 1958
2. 10 CFR Part 100, Appendix A

TOPIC: II-4.D Stability of Slopes

(1) Definition:

Overstressing a slope may cause sudden failure with rapid displacement or shear strain which may damage safety-related structures. The possibility of movement is evaluated by comparing forces resisting failure to those causing failure. An assessment of this ratio should be made to determine the safety factor.

(2) Safety Objective:

To assure that safety-related structures, systems, and components are adequately protected against failure of natural or man-made slopes.

(3) Status:

No work currently being done on this subject for operating plants.

(4) References:

1. Standard Review Plan, Section 2.5.5
2. 10 CFR Part 100, Appendix A
3. Naval Facilities Engineering Command, NAVFAC DM-7, "Design Manual - Soil Mechanics, Foundations, and Earth Structures."

TOPIC: II-4.E Dam Integrity

(1) Definition:

Dam integrity is the ability of a dam to safely perform its intended functions. These functions would normally include remaining stable under all conditions of reservoir operation, controlling seepage to prevent excessive uplifting water pressures or erosion of soil materials, and providing sufficient freeboard and outlet capacity to prevent overtopping.

(2) Safety Objective:

To assure that adequate margins of safety are available under all loading conditions and uncontrolled releases of retained liquid are prevented.

For many projects an important consideration is the necessity of assuring that an adequate quantity of water is available in times of emergency.

(3) Status:

Additional guidance on assuring the integrity of dams is currently being developed by the Office of Standards Development in Regulatory Guide 1.127, "Inspection of Water-Control Structures Associated With Nuclear Power Plants," and through the geotechnical engineering service contract with the U.S. Army Corps of Engineers on design of structures such as ultimate heat sinks.

(4) References:

1. Standard Review Plan, Section 2.5.6
2. 10 CFR Part 100, Appendix A
3. U.S. Army Corps of Engineers, EM 1110-2-1902, "Engineering and Design Stability of Earth and Rock-Fill Dams," Office of Chief of Engineers, 1970
4. U. S. Army Corps of Engineers, EM 1110-2-2300, "Earth and Rock-Filled Dams General Design and Construction Considerations," 1971
5. Regulatory Guide 3.11, "Design, Construction, and Inspection of Embankment Retention Systems for Uranium Mills"

TOPIC: II-4.F Settlement of Foundations and Buried Equipment

(1) Definitions:

Structural loads develop pressures in compressible strata which are not equivalent to the original geostatic pressures. Settlement and differential settlement should be evaluated.

(2) Safety Objective:

To assure that safety-related structures, systems, and components are adequately protected against excessive settlement.

(3) Status:

No work currently being done on this subject for operating plants.

(4) References:

1. Standard Review Plan, Section 2.5.4
2. 10 CFR Part 100, Appendix A
3. Naval Facilities Engineering Command, NAVFAC DM-7, "Design Manual - Soil Mechanics, Foundations, and Earth Structures"

TOPIC: III-1 Classification of Structures, Components, and Systems (Seismic and Quality)

(1) Definition:

Plant structures, systems, and components that are required to withstand the effects of a safe shutdown earthquake and remain functional should be classified as Seismic Category I. Systems and components important to safety should be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. Review the classification of structures, systems, and components important to safety to assure they are of the quality level commensurate with their safety function.

(2) Safety Objective:

To assure that structures, systems, and components will fullfill their intended safety functions in accordance with design requirements. To assure that structures, systems, and components necessary for safety will withstand the effects of the designated safe shutdown earthquake and will remain functional.

(3) Status:

There is currently no Division of Operating Reactors activity to confirm the classification of structures, components, and systems important to safety of operating reactors.

(4) References:

1. Standard Review Plan, Section 3.2.1
2. Standard Review Plan, Section 3.2.2
3. Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants"
4. Regulatory Guide 1.29, "Seismic Design Classification"

TOPIC: III-2 Wind and Tornado Loadings

(1) Definition:

Review the capability of the plant structures, systems, and components to withstand design wind loadings in accordance with 10 CFR 50, Appendix A. The review includes the following: (A) Design Wind Protection; (B) Tornado Wind and Pressure Drop Protection; (C) Effect of Failure of Structures Not Designed for Tornado on Safety of Category I Structures, Systems and Components; (D) Tornado Effects on Emergency Cooling Ponds.

(2) Safety Objective:

To assure that Category I structures, systems, and components are adequately designed for tornado winds and pressure drop, that any damage to structures not designed for tornado-generated forces will not endanger Category I structures, systems, and components, and that tornado winds will not prevent the water in the cooling ponds from acting as a heat sink.

(3) Status:

This review applies to all plants. There are no ongoing reviews concerning this matter.

(4) References:

1. 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 2
2. Standard Review Plan, Sections 3.3, 3.8, and 9.2.5
3. Regulatory Guides
   1.76, "Design Basis Tornado for Nuclear Power Plants"
   1.117, "Protection of Nuclear Plants Against Industrial Sabotage"

TOPIC: III-3.A Effects of High Water Level on Structures

(1) Definition:

If the high water level for the plant is reevaluated and found to be above the original design basis, then review the ability of the plant structures to withstand this water level.

(2) Safety Objective:

To provide assurance that floods or high water level will not jeopardize the structural integrity of the plant seismic Category I structures and that seismic Category I systems and components located within these structures will be adequately protected.

(3) Status:

This review applies to all plants. There are no ongoing reviews concerning this matter.

(4) References:

1. 10 CFR Part 50, Appendix A, GDC 2
2. Standard Review Plan, Sections 2.4, 3.4, and 3.8
3. Regulatory Guides
   1.59, "Design Basis Floods for Nuclear Power Plants"
   1.102, "Flood Protection for Nuclear Power Plants"

TOPIC: III-3.B Structural and Other Consequences (e.g., Flooding of Safety-Related Equipment in Basements) of Failure of Underdrain Systems

(1) Definition:

Some plants rely on underdrain systems to limit the water table elevation at the plant to a safe level. Review underdrain systems of those facilities in which they are used.

(2) Safety Objective:

To assure that the integrity of underdrain systems is maintained because a failure could lead to a rise in water table elevation which, in turn, could jeopardize the integrity of structures or the safety equipment within such structures.

(3) Status:

The structural consequences of the failure of underdrain systems were thoroughly reviewed during the construction-permit review of Douglas Point Units 1 and 2 and Perry Units 1 and 2. There are no ongoing reviews of this topic for operating facilities.

(4) References:

1. 10 CFR Part 50, Appendix A, GDC 2
2. Standard Review Plan, Sections 2.4.13, 3.4, and 3.8

TOPIC: III-3.C Inservice Inspection of Water Control Structures

(1) Definition:

Review the adequacy of the inservice inspection program of water control structures for operating plants to assure conformance with the intent of Regulatory Guide 1.127.

(2) Safety Objective:

To assure that water control structures of a nuclear power facility (for example, dams, reservoirs, and conveyance facilities) are adequately inspected and maintained so as to preclude their deterioration or failure which could result in flooding or in jeopardizing the integrity of the ultimate heat sink for the facility.

(3) Status:

This review applies to all plants. There are no ongoing reviews concerning this matter.

(4) Reference:

Regulatory Guide 1.127, "Inspection of Water-Control Structures Associated With Nuclear Power Plants"

TOPIC: III-4.A Tornado Missiles

(1) Definition:

Plants designed after 1972 have been consistently reviewed for adequate protection against tornadoes. The concern exists, however, that plants reviewed prior to 1972 may not be adequately protected, in particular, those reviewed before 1968 when Atomic Energy Commission criteria on tornado protection were developed.

An assessment of the adequacy of a plant to withstand the impact of tornado missiles would include:

(a) Determination of the capability of the exposed systems, components, and structures to withstand key missiles (including small missiles with penetrating characteristics and larger missiles which result in an overall structural impact),

(b) Determination of whether any areas of the plant require additional protection.

The systems, structures, and components required to be protected because of their importance to safety are identified in Regulatory Guide 1.117.

(2) Safety Objective:

To assure that those structures, systems, and components necessary to ensure:

(a) The integrity of the reactor coolant pressure boundary,

(b) The capability to shut down the reactor and maintain it in a safe shutdown condition, and

(c) The capability to prevent accidents which could result in unacceptable offsite exposures,

can withstand the impact of an appropriate postulated spectrum of tornado-generated missiles.

(3) Status:

The Regulatory Requirements Review Committee (RRRC) has approved case-by-case rereviews of plants against criteria in Regulatory Guide 1.117, which establishes the systems, structures, and components required to be protected against tornado missiles. This rereview was deferred pending the formation of the SEP.

The RRRC is in the process of rereviewing Standard Review Plan, Section 3.5.1.4, which establishes appropriate missiles and impact velocities for new applications.

Electric Power Research Institute (EPRI) has missile research in progress.

(4) References:

1. Standard Review Plan, Section 3.5.1.4
2. Regulatory Guide 1.117, "Tornado Design Classification"

TOPIC: III-4.B Turbine Missiles

(1) Definition:

A number of nonnuclear plants and one nuclear plant (Shippingport) have experienced turbine disk failures. Rancho Seco has had chemistry problems leading to sodium deposits which caused stress-corrosion cracking of disks. Failure of turbine disks and rotors can result in high energy missiles which have the potential for resulting in plant releases in excess of 10 CFR 100 exposure guidelines.

Two areas of concern should be considered:

(a) Design overspeed failures - material quality of disk and rotor, inservice inspection for flaws, chemistry conditions leading to stress-corrosion cracking, and

(b) Destructive overspeed failures - reliability of electrical overspeed protection system, reliability and testing program for stop and control valves, inservice inspection of valves.

The focus of the review would be on turbine disk integrity and overspeed protection, including stop, intercept, and control valve reliability.

(2) <u>Safety Objective</u>:

To assure that all the structures, systems, and components important to safety (identified in Regulatory Guide 1.117) have adequate protection against potential turbine missiles either by structural barriers or a high degree of assurance that failures at design (120%) or destructive (180%) overspeed will not occur.

(3) <u>Status</u>:

No work currently being done on this subject for operating plants. Electric Power Research Institute (EPRI) has missile research in progress.

(4) <u>References</u>:

1. Regulatory Guides
   1.115, "Protection Against Low Trajectory Turbine Missiles"
   1.117, "Tornado Design Classification"
2. Standard Review Plan, Section 3.5.1.3

<u>TOPIC</u>: III-4.C Internally Generated Missiles

(1) <u>Definition</u>:

Review the probability of missile generation and the extent to which safety-related structures, systems, and components are protected against the effects of potential internally generated missiles (including missiles generated inside or outside the containment).

(2) <u>Safety Objective</u>:

To provide assurance that the integrity of the safety-related structures, systems, and components will not be impaired and that they may be relied on to perform their safety functions following any postulated internally generated missile.

(3) <u>Status</u>:

No work currently being done on this subject for operating plants. Electric Power Research Institute (EPRI) has missile research in progress.

(4) Reference:

Standard Review Plan, Sections 3.5.1.1 and 3.5.1.2

TOPIC: III-4.D Site-Proximity Missiles (Including Aircraft)

(1) Definition:

Review the extent to which safety-related structures, systems, and components are protected against the effects of missiles postulated in Topic II-1.C, including postulated aircraft crashes and resulting fires.

(2) Safety Objective:

To provide assurance that the integrity of the safety-related structures, systems, and components will not be impaired and that they will perform their safety functions in the event of a site-proximity missile.

(3) Status:

No work currently being done on this subject for operating plants. Electric Power Research Institute has missile research in progress.

(4) Reference:

Standard Review Plan, Sections 3.5.1.5, 3.5.1.6, 3.5.2, and 3.5.3

TOPIC: III-5.A Effects of Pipe Break on Structures, Systems, and Components Inside Containment

(1) Definition:

Review the licensee's break and crack location criteria and methods of analysis for evaluating postulated breaks and cracks in high and moderate energy fluid system piping inside containment. The review includes consideration of compartment pressurization, pipe whip, jet impingement, environmental effects, and flooding. Regulatory Guide 1.46 does not require that cracks be postulated inside containment. However, the recent proposed revision to Standard Review Plan, Section 3.6.2, "Determination of Break Locations and Dynamic Effects Associated With the Postulated Rupture of Piping," recommends that cracks be postulated inside containment. Old and current plants are not postulating cracks.

(2) Safety Objective:

To assure that the integrity of structures, systems, and components relied upon for safe reactor shutdown or to mitigate the consequences of a postulated pipe break is maintained.

(3) Status:

This program has not been started for facilities licensed prior to about early 1974. Subsequent to that date, this topic was included in the operating-license review and has been completed for later facilities.

(4) References:

1. 10 CFR Part 50, Appendix A, GDC 4
2. American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code," Section III
3. Standard Review Plan, Sections 3.6.2 and 3.8
4. Regulatory Guides
   1.46, "Protection Against Pipe Whip Inside Containment"
   1.29, "Seismic Design Classification"

TOPIC: III-5.B Pipe Break Outside Containment

(1) Definition:

Review the licensee's break and crack location criteria and methods of analysis for evaluating postulated breaks and cracks in high and moderate energy fluid system piping located outside containment. The review includes consideration of compartment pressurization, pipe whip, jet impingement, environmental effects, and flooding.

(2) Safety Objective:

To assure that pipe breaks would not cause the loss of needed functions of safety-related systems, structures, and components and to assure that the plant can be safely shut down in the event of such breaks.

(3) Status:

This task is complete for all operating plants with the exception of three plants for which the review is in progress.

(4) References:

1. 10 CFR Part 50, Appendix A, GDC 4
2. American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code," Section III
3. Standard Review Plan, Section 3.6.1
4. Regulatory Guides
   1.46, "Protection Against Pipe Whip Inside Containment"
   1.29, "Seismic Design Classification"
5. Standard Review Plan, Branch Technical Position MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment"
6. NUREG-0328, "Regulatory Licensing: Status Summary Report," (Pink Book) Issue 3-25
7. Standard Review Plan, Section 3.6.2

TOPIC: III-6 Seismic Design Considerations

(1) Definition:

Review and evaluate the original plant design criteria in the following areas: Seismic Input, Analysis and Design Criteria, Qualification of Electrical and Mechanical Equipment, Seismic Instrumentation, Seismic

Categorization, and the effect of failure of non-Category I structures on the safety of Category I structures, systems, and components.

(2) Safety Objective:

To ensure the capability of the plant to withstand the effect of earthquakes.

(3) Status:

Humboldt Bay and San Onofre plants are currently undergoing seismic review. Technical Assistance Contracts:

(a) Seismic Conservatism (Lawrence Livermore Laboratory)
(b) Elasto-Plastic Seismic Analysis (Lawrence Livermore Laboratory)
(c) Seismic Review of Operating Plants (Newmark)

(4) References:

1. Standard Review Plan, Sections 2.5, 3.7, 3.8, 3.9, and 3.10
2. Regulatory Guides
   1.12, "Instrumentation for Earthquakes"
   1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants"
   1.61, "Damping Values for Seismic Design of Nuclear Power Plants"
   1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis"
   1.122, "Development of Flood Design Spectra for Seismic Design of Floor-Supported Equipment or Components"

TOPIC: III-7.A Inservice Inspection, Including Prestressed Concrete Containments With Either Grouted or Ungrouted Tendons

(1) Definition:

Review licensee's inspection program for all Category I structures including steel, reinforced concrete, and prestressed concrete containments. The program should include investigations for possible corrosion and cracking of steel containments, excessive cracking of concrete structures, lift-off tests of tendons, periodic testing of prestressing tendons for containments with grouted tendons, and possible deterioration of prestressed containments.

(2) Safety Objective:

To assure that the licensee's inspection program will detect any damaging deterioration of the structures and that they will be capable of performing as required by 10 CFR 50, Appendix A.

(3) Status:

This review applies to all plants. There are no ongoing reviews concerning this matter.

(4) References:

1. 10 CFR Part 50, Appendix A
2. Standard Review Plan, Section 3.8
3. Regulatory Guides
   1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures"
   1.90, "Inservice Inspection of Prestressed Concrete Containment Structures With Grouted Tendons"

TOPIC: III-7.B Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria

(1) Definition:

Review the design codes, design criteria, and load combinations for all Category I structures (that is, containment, structures inside containment, and structures outside containment).

(2) Safety Objective:

To provide assurance that the plant Category I structures will withstand the NRC specific design conditions without impairment or structural integrity or the performance of required safety functions.

(3) Status:

This review applies to all plants. There are no ongoing reviews concerning this matter.

(4) References:

1. 10 CFR Part 50, Appendix A, GDC 2 and 4
2. Standard Review Plan, Section 3.8

TOPIC: III-7.C Delamination of Prestressed Concrete Containment Structures

(1) Definition:

Review the design of prestressed concrete containment structures to assess the likelihood of delamination occurring in the shell walls or dome and to evaluate the consequences, if any.

(2) Safety Objective:

To assure that the licensee's design and construction methods have provided a structure which will maintain its integrity and will perform its intended function. Delaminations (internal cracking of concrete in planes roughly parallel to the surface) could possibly reduce the capability of the concrete to withstand compression.

(3) Status:

This review applies to all plants with prestressed concrete containments. A delamination occurred in the domes of the Turkey Point and Crystal River prestressed concrete containments. No evidence of such occurrences have been reported at other plants; however, no specific inspections have been made for any delaminations. It is not clear if the Structural Integrity Test or the existing inservice inspection programs would discover the existence of any delaminations.

(4) References:

Safety Evaluation Reports for Turkey Point (Docket No. 50-250/251) and Crystal River (Docket No. 50-302)

TOPIC: III-7.D Containment Structural Integrity Tests

(1) Definition:

Review the licensee's structural integrity testing procedure to ensure compliance with the requirements of 10 CFR 50, Appendix A.

(2) Safety Objective:

To assure that the licensee's design and constructive methods provide a structure which will safely perform its intended functions.

(3) Status:

This review applies to all plants. To our knowledge, all containments have had a structural integrity test. This opinion should be verified.

(4) References:

1. 10 CFR Part 50, Appendix A
2. Standard Review Plan, Sections 3.8.1 and 3.8.2

TOPIC: III-8.A Loose-Parts Monitoring and Core Barrel Vibration Monitoring

(1) Definition:

Inservice surveillance programs to detect loose parts and excessive motion of the main core support structure.

(2) Safety Objective:

To detect loose parts or excessive vibration before they can cause flow blockage or mechanical damage to the fuel or other safety-related components.

(3) Status:

The NRC staff currently requires applicants to describe and licensees to implement a loose-part detection program. Guidance for such a program is provided in a newly proposed Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors." The regulatory guide outlines the minimum system characteristics which the NRC staff feels are necessary for a workable system and combines this with a technical specification and reporting procedures for a complete and enforceable loose-part detection program.

The concept of detecting core barrel motion through use of excore neutron detectors is well established. A proposed regulatory guide that describes an acceptable core barrel vibration monitoring program has been temporarily placed on "hold" to permit the NRC staff and its consultants (Oak Ridge National Laboratory Inspection and Enforcement Group) time to evaluate apparently anomalous data from core barrel motion monitoring programs that are currently in service as part of the technical specification requirements for certain licensees.

(4) References:

1. Combustion Engineering, CE Report CEN-5(P), "Palisades Reactor Internals Wear Report," March 1, 1974
2. Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors"

TOPIC: III-8.B Control Rod Drive Mechanism Integrity

(1) Definition:

Review and evaluate the reliability, operability and any reported mechanical failures in control rod drives.

(2) Safety Objective:

To assure that the integrity and operability of control rod drives is adequately maintained so that they will be capable of normal reactor control and prompt reactor shutdown, if required.

(3) Status:

The Division of Operating Reactors Engineering Branch is currently evaluating the failure modes and internal component redesigns of BWR control rod drives to preclude stress corrosion and thermal fatigue cracking. There have been no reported generic failures of PWR drives.

(4) Reference:

General Electric, NEDO-21021, "Test Program for Collet Retainer Tube," June 23, 1976.

TOPIC: III-8.C Irradiation Damage, Use of Sensitized Stainless Steel, and Fatigue Resistance

(1) Definition:

Review the safety aspects that affect reactor vessel internals integrity for compliance with 10 CFR Part 50, including radiation damage, use of sensitized stainless steel, and fatigue resistance.

(2) Safety Objective:

To assure continued reactor vessel internals integrity and compliance with 10 CFR Part 50 and applicable industry Codes and Standards.

(3) Status:

The Engineering Branch, Division of Operating Reactors, currently has no review programs relating to reactor vessel internals integrity.

(4) References:

1. 10 CFR Part 50, Appendix A
2. American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code," Section III
3. American Society of Testing Materials, ASTM A-262-70, "Standard Recommended Practices for Detecting Susceptibility to Intergranular Attack in Stainless Steels"
4. Regulatory Guides
   1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants"
   1.44, "Control of the Use of Sensitized Stainless Steel"
   1.61, "Damping Values for Seismic Design of Nuclear Power Plants"

TOPIC: III-8.D Core Supports and Fuel Integrity

(1) Definition:

Abnormal loading conditions on the core supports and fuel assemblies due to seismic events or loss-of-coolant accidents (LOCAs) could cause fuel damage due to impact between fuel assemblies and upper- and lower-grid plates or lateral impact between fuel assemblies and the core baffle wall. The resulting damage could result in loss of coolable heat transfer geometry, make it impossible to insert control rods, or cause releases of radioactive materials due to fuel pin failure.

(2) Safety Objective:

To assure that all credible loading conditions on core supports and fuel assemblies will not result in unacceptable fuel damage or distortion.

(3) Status:

The Division of Operating Reactors is currently reviewing the dynamic loads imposed on the fuel assemblies during a LOCA. Independent analyses are being conducted by staff consultants.

(4) Reference:

American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code," Section III

(5) Basis for Deletion (Related TMI Task, USI, or Other SEP Topic):

USI A-2, "Asymmetric Blowdown Loads on Reactor Primary Coolant System" (NUREG-0649)

USI A-2 requires that an analysis be performed by licensees to assess the design adequacy of the reactor vessel supports and other structures to withstand the loads when asymmetric LOCA forces are taken into account. The staff has completed its investigation and concluded that an acceptable basis has been provided in NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems," January 1981, for performing and reviewing plant analyses for asymmetric LOCA loads. The structural acceptance criteria specified in NUREG-0609 are as follows:

The structural integrity of the primary system including the reactor pressure vessel, reactor pressure vessel internals, primary coolant loop, and components must be evaluated against appropriate acceptance criteria to determine if acceptable margins of safety exist. Allowable limits and appropriate loading combinations are set forth in Standard Review Plans (SRPs), which are listed in the table that follows. The staff recognizes that in some specific cases where "as-built" designs are being reevaluated for asymmetric LOCA loads, these design limits may be exceeded. Acceptance of alternative allowable limits will be based on a case-by-case evaluation of the safety margins.

Load-combination criteria in general were not addressed as part of this study. Currently the staff requires that seismic and LOCA response be combined, along with responses due to other loading as specified by the SRP. An acceptable method for combining elastically generated seismic and LOCA responses is provided in NUREG-0484. Acceptable methods for combining response generated by an inelastic LOCA analysis and elastic seismic analyses will be evaluated on a case-by-case basis.

Since USI A-2 also requires the investigation of seismic and LOCA response be combined, the evaluation required by USI A-2 is identical to SEP Topic III-8.D; therefore, this SEP topic has been deleted.

Item	SRP Section
Reactor pressure vessel	3.9.3
Reactor internals	3.9.5, 3.9.1
Primary coolant loop piping	3.9.3
ECCS piping	3.9.3
RPV, SG, pump supports	3.8.3
Biological shield wall	3.8.3
Steam-generator compartment wall	3.8.3
Neutron-shield tank	3.8.3

TOPIC: III-9 Support Integrity

(1) Definition:

Review the design, design loads, and materials integrity including corrosion and fracture toughness and the inservice inspection programs of supports and restraints including bolting for the reactor vessel, steam generator, reactor coolant pump, torus, and other Class 1, 2, and 3 safety-related components and piping systems.

(2) Safety Objective:

To assure adequate support and/or restraint of safety-related systems and components under normal and accident loads so that they will not be prevented from performing their intended functions because of support failures.

(3) Status:

The Division of Operating Reactors has ongoing programs to review component supports. Current emphasis is on primary system supports and on piping system supports and restraints (snubbers).

(4) References:

1. American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code," Section III
2. NUREG-0328, "Regulatory Licensing: Status Summary Report" (Pink Book), Generic Topics 3-5 and 3-43

(5) Basis for Deletion (Related TMI Task, USI, or other SEP Topic):

(a) USI A-12, "Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports" (NUREG-0510 and NUREG-0606)

The original scope of USI A-12 was the review of the steam generator and reactor coolant pump supports of pressurized water reactors.

However, the staff has expanded the review to include other support structures, such as boiling water reactor (BWR) vessel supports, BWR pump supports, pressurized water reactor (PWR) vessel supports and PWR pressurizer supports (NUREG-0577, Section 1.3). This expanded review will be undertaken in accordance with the guidance of Section 4 of NUREG-0577.

(b) USI A-7, "MARK I Containment Long-Term Program" (NUREG-0649)

Support integrity of the torus is being evaluated under USI A-7. Under this task, a short-term program that evaluated Mark I containment has provided assurance that the Mark I containment system of each operating BWR facility would maintain its integrity and functional capability during a postulated loss-of-coolant accident. A longer term program for BWR facilities, not yet licensed, is planned wherein the NRC staff will evaluate the loads, load combinations, and associated structural acceptance criteria proposed by the Mark I Owners Group prior to the performance of plant-unique structural evaluations. The Mark I Owners Group has initiated a comprehensive testing and evaluation program to define design-basis loads for the Mark I containment system and to establish structural acceptance criteria which will assure margins of safety for the containment system which are equivalent to that which is currently specified in the ASME Boiler and Pressure Vessel Code. Also included in their program is an evaluation of the need for structural modifications and/or load mitigation devices to assure adequate Mark I containment system structural safety margins.

(c) USI A-24, "Qualification of Class 1E Safety-Related Equipment" (NUREG-0371 and NUREG-0606)

Snubber operability and degradation of seals are covered under USI A-24.

(d) USI A-46, "Seismic Qualification of Equipment in Operating Plants" (NUREG-0705)

Mechanical snubbers are covered under USI A-46.

(e) SEP Topic III-6, "Seismic Design Considerations"

Snubbers are evaluated for capacity under SEP Topic III-6.

(f) SEP Topic V-1, "Compliance With Codes and Standards (10 CFR 50.55a)"

Inservice inspection requirements for supports are covered under SEP Topic V-1, which refers to 10 CFR 50.55a. SEP plants currently have surveillance Technical Specifications on snubbers.

The evaluation required by USI A-12, A-7, A-24, and A-46 and SEP Topics III-6 and V-1 is identical to the evaluation required by SEP Topic III-9; therefore, this SEP topic has been deleted.

TOPIC: III-10.A Thermal-Overload Protection for Motors of Motor-Operated Valves

(1) Definition:

The primary objective of thermal overload relays is to protect motor windings of motor-operated valves (MOVs) against excessive heating. This feature of thermal overload relays could, however, interfere with the successful functioning of a safety-related system. In nuclear plant safety system application, the ultimate criterion should be to drive the valve to its proper position to mitigate the consequences of an accident, rather than to be concerned with degradation or failure of the motor due to excess heating.

(2) Safety Objective:

To assure that (1) thermal overload protection, if provided for MOVs, should have the trip setpoint at a value high enough to prevent spurious trips due to design inaccuracies, trip setpoint drift, or variation in the ambient temperature at the installed location; (2) the circuits which bypass the thermal overload protection under accident conditions should be designed to IEEE Std. 279-1971 criteria, as appropriate for the rest of the safety-related system; and (3) in MOV designs that use a torque switch instead of a limit switch to limit the opening or closing of the valve, the automatic opening or closing signal should be used in conjunction with a corresponding limit switch and thermal overload should remain as backup protection.

(3) Status:

The staff position (Reference 1) is implemented on designs of new applications (construction permit and operating license).

(4) References:

1. Standard Review Plan, Branch Technical Position EICSB 27, "Design Criteria for Thermal Overload Protection for Motors of Motor-Operated Valves"
2. Institute of Electrical and Electronics Engineers, IEEE Std. 279-1971, Criteria for Protection System for Nuclear Power Generating Stations"
3. Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves"

TOPIC: III-10.B Pump Flywheel Integrity

(1) Definition:

Review the PWR reactor coolant pump flywheel inservice inspection programs of operating plants to assure that they comply with the intent of Regulatory Guide 1.14 and review reports of flywheel flaws if found by inservice inspections. (BWR reactor coolant pumps do not have flywheels.)

(2) Safety Objective:

To assure that pump flywheel integrity is maintained to prevent failure at normal operating speeds and at speeds that might be reached under accident conditions and thus preclude the generation of missiles.

(3) Status:

The inservice inspection programs for flywheels of older PWRs have not been reviewed for compliance with the intent of Regulatory Guide 1.14.

(4) Reference:

Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity"

TOPIC: III-10.C Surveillance Requirements on BWR Recirculation Pumps and Discharge Valves

(1) Definition:

At facilities which have completed the low pressure coolant injection system (LPCIS) modification, the recirculation pump discharge valves and bypass valves are now required to close upon initiation of LPCIS. The closure of these discharge valves is necessary to isolate a pipe break in a suction line to prevent loss of cooling water by reverse flow through the recirculation pump or its bypass line and out the break.

(2) Safety Objective:

To assure effective core cooling in the event of a BWR recirculation line break on the pump suction line by closing the pump discharge valve and bypass line valve.

(3) Status:

All licensees of facilities with completed LPCIS modification have been sent letters requesting that they apply for a license amendment to incorporate technical specification surveillance requirements on recirculation pump discharge valves and bypass valves. New BWRs have the LPCIS modification and technical specification surveillance requirements.

(4) Reference:

NUREG-0328, "Regulatory Licensing: Status Summary Report," (Pink Book) Issue 3-46, June 17, 1977

TOPIC: III-11 Component Integrity

(1) Definition:

Review licensee's criteria, testing procedures, and dynamic analyses employed to assure the structural integrity and functional operability of safety-related mechanical equipment under faulted conditions and accident loads. Included are mechanical equipment such as pumps, valves, fans, pump drives, heat exchanger tube bundles, valve actuators, battery and instrument racks, control consoles, cabinets, panels, and cable trays.

(2) Safety Objective:

To confirm the ability of safety-related mechanical equipment having experienced problems to function as needed during and after a faulted or accident condition. The capability of safety-related mechanical equipment to perform necessary protective actions is essential for plant safety.

(3) Status:

This review is not currently under way in the Divisions of Operating Reactors.

(4) References:

1. 10 CFR Part 50, Section 50.55a
2. 10 CFR Part 50, Appendix A, GDC 2, 4, 14, and 15
3. Standard Review Plan, Section 3.9.2
4. American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code," Section III,
5. Regulatory Guides
   1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing"
   1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants"
6. Institute of Electrical and Electronics Engineers, IEEE Std. 344-1975, "Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations"
7. Standard Review Plan, Section 3.9.3

(5) Basis for Deletion (Related TMI Task, USI, or Other SEP Topic):

(a) USI A-46, "Seismic Qualification of Equipment in Operating Plants" (NUREG-0606 and NUREG-0705)

The component integrity (both structural integrity and functional operability) for safety-related mechanical and electrical equipment for all operating plants including SEP plants will be addressed in this new USI (A-46).

(b) USI A-2, "Asymmetric Blowdown Loads on Reactor Primary Coolant System" (NUREG-0649)

The assessment of faulted loads for the primary loop is being performed under USI A-2. Furthermore, the assessment of high-energy pipe breaks considers the effect of accident loads with regard to jet impingement, pipe whip, and other reaction loads.

(c) SEP Topic III-6, "Seismic Design Considerations"

The evaluation of equipment structural integrity under seismic loads will be performed under SEP Topic III-6.

The evaluations required by USI A-46 and A-2 and SEP Topic III-6 are identical to SEP Topic III-11; therefore, this SEP topic has been deleted.

TOPIC: III-12 Environmental Qualification of Safety-Related Equipment

(1) Definition:

Safety-related electrical and mechanical equipment that is required to survive and function under environmental conditions calculated to result from a loss-of-coolant accident (LOCA) or a postulated main steam line break accident inside containment must be environmentally qualified. In addition, determine whether environment-induced failures of nonsafety-related equipment could interfere with the operation of safety equipment. Special attention should be given to the effect of beta radiation on exposed organic surfaces, such as gaskets.

(2) Safety Objective:

To assure that the mechanical and Class IE electrical equipment of safety systems has been qualified for the most severe environment (temperature, pressure, humidity, chemistry, and radiation) of design basis accidents.

(3) Status:

Westinghouse is conducting a verification program which is expected to be completed by the end of 1977 for those plants qualified to IEEE 323-1971. The Office of Nuclear Regulatory Research is sponsoring programs relating to Class IE equipment qualification, the results of which can be utilized to determine the adequacy of the equipment previously qualified.

(4) References:

1. NUREG-0153, "Staff Discussion of Twelve Additional Technical Issues Raised by Responses to November 3, 1976 Memorandum From Director, NRR, to NRR Staff," Issue 25, "Qualification of Safety-Related Equipment," December 1976
2. Division of Operating Reactors, DOR Technical Activities, Category B, Item 34, "Environmental Qualifications of Safety-Related Equipment (Post LOCA)," May 1977
3. Division of Systems Safety, DSS Technical Activities, Category A, Item 33, "Qualification of Class IE Safety-Related Equipment," April 1977
4. Regulatory Guide 1.89, "Qualification of Class IE Equipment for Nuclear Power Plants"

(5) Basis for Deletion (Related TMI Task, USI, or other SEP Topic):

USI A-24, "Qualification of Class IE Safety-Related Equipment" (NUREG-0371 and NUREG-0606)

The issue identified in Reference 1 (NUREG-0153, Item 25) and the review criteria, that is, Regulatory Guide 1.89, are identical to those specified in USI A-24. The Task Action Plan for USI A-24

(NUREG-0371) covers the environmental qualification of both electrical and mechanical safety-related equipment.

The evaluation required by USI A-24 is identical to SEP Topic III-12; therefore, this SEP topic has been deleted.

TOPIC: IV-1.A Operation With Less Than All Loops in Service

(1) Definition:

A number of BWR and PWR licensees have requested authorization to operate with one of the recirculation loops (BWR) or steam generator loops (PWR) out of service. These proposals are being reviewed generically with regard to analytical methods. Plant-specific reviews will be done to determine appropriate Technical Specification limits. Plant-specific reviews will address results of LOCA analyses using generically approved methods. Analysis of accidents (other than LOCA) and operating transients resulting from operation in the (N-1) loop mode have been reviewed on a "lead plant basis." Most of this effort has been completed. Tests have been conducted by General Electric which show that significant core flow asymmetries do not exist with single-loop operation for two-loop plants; however, there is backflow through inactive jet pumps. Therefore, for single-loop operation, modifications are necessary in trip settings which take inputs from jet pump drive flow. These will be determined on a plant-specific basis.

(2) Safety Objective:

To provide assurance that operation with less than all coolant loops in operation will not result in decreased safety margins.

(3) Status:

A combination of generic and plant-specific reviews is being performed on both BWRs and PWRs.

TOPIC: IV-2 Reactivity Control Systems Including Functional Design and Protection Against Single Failures

(1) Definition:

General Design Criterion 25 requires that the reactor protection system be designed to assure that fuel-damage limits are never exceeded in the event of any single failure of the reactivity control systems. Reactivity control systems need not be designed single failure proof, but the protection system (which is designed against single failures) should be capable of limiting fuel damage in the event of a reactivity control system single failure.

(2) Safety Objective:

To assure that for all credible reactivity control system failures, the protection system will limit fuel damage to acceptable limits.

(3) Status:

NRC has concluded that revisions to existing licenses are not warranted. Staff effort on this issue will continue at a low level.

(4) References:

1. NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum From Director, NRR, to NRR Staff," Issue No. 6, "Protection Against Single Failures in Reactivity Control Systems," December 1976.
2. Standard Review Plan, Section 15.4.3

TOPIC: IV-3 BWR Jet Pump Operating Indications

(1) Definition:

If a jet pump BWR operates with a failed jet pump, it may be impossible to reflood the core in the event of a LOCA. Some BWRs have experienced jet pump instrument sensing line failures. With a sensing line failed, it may not be possible to accurately measure core flow or to detect failure of a jet pump.

(2) Safety Objective:

To assure that the core flow can be determined. Also to assure the ability to detect a jet pump failure for a range of crack/break sizes at various locations on the pump.

(3) Status:

This issue is currently being reviewed for Dresden Units 2 and 3 and Quad Cities Units 1 and 2. The topic has generic implications for all jet pump BWR plants.

(4) References:

1. Letters from Commonwealth Edison Company to NRC, dated September 19, 1975, March 3, 1976, and June 7, 1976.
2. Letter from NRC to Commonwealth Edison Company, dated January 19, 1976.
3. Memorandum from J. H. Sniezek, NRC, to D. L. Ziemann, dated November 19, 1975.

TOPIC: V-1 Compliance With Codes and Standard (10 CFR 50.55a)

(1) Definition:

Review the licensee's inservice inspection and testing programs for Class 1, 2, and 3 pressure vessels, piping, pumps and valves and other safety-related components to assure compliance with the American Society of Mechanical Engineers (ASME) Code, Sections III and XI, as required by 10 CFR 50.55a. This review will also include review of the inservice inspection and testing program applicable to isolation condensers of the early operating BWRs.

(2) Safety Objective:

To assure that the initial integrity of components is maintained throughout service life.

(3) Status:

NUREG-0081 was completed for reactor vessels not designed to ASME Code, Section III. The Engineering Branch conducts a generic review of all plants for compliance with inspection requirements of 10 CFR 50.55a(g) and fracture toughness requirements of 10 CFR 50.55a(i). This program will continue for the life of operating reactors.

(4) References:

1. 10 CFR Part 50, Section 50.55a
2. American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code," Sections III and XI
3. NUREG-0081, "Evaluation of the Integrity of Reactor Vessels Designed to ASME Code, Section I and/or VIII," July 1976
4. Memorandum from V. Stello, NRC, to B. H. Grier, October 12, 1976

TOPIC: V-2 Applicability of Code Cases

(1) Definition:

Review Code Cases currently accepted by the NRC, as indicated in Regulatory Guides 1.84 and 1.85.

(2) Safety Objective:

To assure that only those Code Cases which are acceptable to the NRC are utilized by the licensee in the design, fabrication, or repair of the plant. The use of Code Cases other than those contained in Regulatory Guides 1.84 and 1.85 are addressed on a case-by-case basis to assess their acceptability.

(3) Status:

The Engineering Branch, Division of Operating Reactors, routinely reviews design modifications and component repairs (for example, reactor vessel nozzles) to assure compliance with NRC acceptable Code Cases. The program is ongoing on an as-needed basis.

(4) References:

Regulatory Guides
1.84, "Design and Fabrication Code Case Acceptability - ASME Section III, Division 1"
1.85, "Materials Code Case Acceptability - ASME Section III, Division 1"

TOPIC: V-3 Overpressurization Protection

(1) Definition:

Inadvertent overpressurization of the primary system at temperatures below the nil ductility transition temperature may result in reactor vessel failure during heatup and pressurization. Such overpressure transients are caused by pressure surges when the primary system is water solid. The most severe transients have occurred when a charging pump starts up or inadvertent closing of a letdown valve with a charging pump running. Pressure temperature limits as a function of neutron fluence of the material at the reactor vessel beltline are specified in 10 CFR 50, Appendix G. All PWR licensees have been directed to institute interim administrative procedures to prevent damaging pressure transients and on a longer time scale to provide permanent protection which will probably include hardware changes such as high-capacity safety relief valves.

(2) Safety Objective:

To protect the primary system from potentially damaging overpressurization transients during plant pressurization and heatup.

(3) Status:

Generic review of all PWR licensee submittals is under way. Criteria for evaluation have been developed and refined by the Office of Nuclear Reactor Regulation and the Office of Nuclear Regulatory Research. An effort is being made to complete the review sufficiently early to ensure installation of mitigating systems by the end of 1977.

(4) Reference:

NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum From Director, NRR to NRR Staff," November 1976

(5) Basis for Deletion (Related TMI Task, USI, or Other SEP Topic):

USI A-26, "Reactor Vessel Pressure Transient Protection" (NUREG-0410)

Under USI A-26, licensees were requested to modify their systems and procedures to protect against low temperature overpressurization. All operating PWRs have made these modifications, and safety evaluation reports for the SEP plants have been issued.

The evaluation required by USI A-26 is identical to SEP Topic V-3; therefore, this SEP topic has been deleted.

TOPIC: V-4 Piping and Safe-End Integrity

(1) Definition:

Review the safety aspects that affect BWR and PWR piping and safe-end integrity for compliance with 10 CFR Part 50, including fracture toughness, flaw evaluation, stress corrosion cracking in BWR and PWR piping, and control of materials and welding.

(2) Safety Objective:

To ensure continued piping integrity and compliance with 10 CFR Part 50 and applicable industry codes and standards.

(3) Status:

The Engineering Branch, Division of Operating Reactors, is conducting an ongoing program that includes the as-needed review of those aspects necessary to ensure the continuing integrity of piping systems important to safety including stress corrosion cracking of BWR coolant pressure boundary piping. This program will continue for the life of operating reactors.

(4) Reference:

American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code," Section XI

(5) Basis for Deletion (Related TMI Task, USI, or Other SEP Topic):

(a) USI A-42, "Pipe Cracks in Boiling Water Reactors" (NUREG-0510)

The scope of USI A-42 is the study of stress corrosion cracking in BWR piping. NUREG-0313, Revision 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," is the resolution of USI A-42 and presents staff positions.

(b) USI A-10, "BWR Feedwater Nozzle Cracking and Control Rod Drive Hydraulics Return Line Nozzle Cracking" (NUREG-0649)

(c) NRR Generic Activity C-7, "PWR System Piping" (NUREG-0471)

The scope of this activity is the study of stress corrosion cracking in PWR piping. NUREG-0691, "Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors," recommends the same corrective actions (pp. 2-12) proposed for BWRs in NUREG-0313, Revision 1, USI A-42.

The evaluation required by USI A-42 and Task C-7 is identical to the evaluation required by SEP Topic V-4; therefore, this SEP topic has been deleted.

TOPIC: V-5 Reactor Coolant Pressure Boundary (RCPB) Leakage Detection

(1) Definition:

Reactor primary coolant leakage detection systems are a significant means of preventing primary system boundary failure by identifying leaks before failures occur.

(2) Safety Objective:

To provide reliable and sensitive leakage detection systems to identify primary system leaks at an early stage before failures occur.

(3) Status:

This issue has been resolved for all plants which have recently received an operating license by requiring conformance to Regulatory Guide 1.45. Individual older plants have not been systematically reviewed and leakage detection systems may need upgrading on a plant-by-plant basis.

(4) References:

1. Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems"
2. Standard Review Plan, Section 5.2.5

TOPIC: V-6 Reactor Vessel Integrity

(1) Definition:

Review the safety aspects that affect BWR and PWR reactor vessel and nozzle integrity for compliance with 10 CFR Part 50, including fracture toughness, neutron irradiation, evaluation of surveillance programs, operating limitations, inservice inspection and flaw evaluation, and transient analyses.

(2) Safety Ojective:

To assure continued reactor vessel integrity and compliance with 10 CFR Part 50 and applicable industry codes and standards.

(3) Status:

The Engineering Branch, Division of Operating Reactors, is conducting ongoing programs that include the periodic review of aspects necessary to ensure the continued integrity of reactor vessels. These programs include BWR feedwater and control rod drive nozzle cracking, low upper-shelf toughness, radiation effects, reactor vessel materials surveillance, and updating of operating plants' inservice inspection programs and will continue for the life of operating reactors.

(4) References:

1. NUREG-0312, "Interim Technical Report on BWR Feedwater and Control Rod Drive Return Line Nozzle Cracking," July 1977
2. 10 CFR Part 50, Appendix G
3. Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials"
4. American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code," Section III, Appendix G
5. American Society of Testing Materials, ASTM E185, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels"

6. American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code," Section XI
7. NUREG-0328, "Regulatory Licensing: Status Summary Report" (Pink Book), Issue 3-9, 3-21, 3-41

TOPIC: V-7 Reactor Coolant Pump Overspeed

(1) Definition:

Review the potential for reactor coolant pumps to fail because of overspeed in the unlikely event of a major loss-of-coolant accident (LOCA).

(2) Safety Objective:

To assure that, in the event of a major LOCA, a reactor coolant pump assembly is not driven to a speed which would cause structural failure of the unit and result in missiles which could increase the consequences of the LOCA. Of greatest concern are the PWR pump flywheels because of their mass and rotational energy.

(3) Status:

An indepth review of this topic was performed by the Atomic Energy Commission staff and reported to the Advisory Committee on Reactor Safeguards (ACRS) in 1973 (Reference 1). The staff concluded that, because of the small likelihood for the occurrence of a pump overspeed event that could seriously increase the consequences resulting from a LOCA (less than 10-8 per plant year), the action taken by the staff to assess this problem in a generic fashion outside the context of individual application reviews is an acceptable course to follow. A generic experimental program to be completed in 1978 by the Electric Power Research Institute is expected to provide data to verify pump model overspeed predictions.

(4) References:

1. Letter from R. C. DeYoung, NRC, to Harold G. Mangelsdorf, ACRS, August 6, 1973, transmitting "Report on Reactor Coolant Pump Overspeed During a LOCA," August 3, 1973.
2. Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity"

TOPIC: V-8 Steam Generator (SG) Integrity

(1) Definition:

Review the safety aspects affecting operation of steam generators including secondary water chemistry, tube plugging criteria, inservice inspection, possibly including a dimensional inspection for proper evaluation of denting, steam generator tube leakage, tube denting, flow-induced vibration of steam generator tubes, tube repair, and tube bundle or steam generator replacement.

(2) Safety Objective:

To ensure that acceptable levels of integrity of that portion of the reactor coolant pressure boundary made up by the steam generator are maintained in accordance with current codes, standards, and/or regulatory criteria during normal and postulated accident conditions. The integrity of the steam generator is needed to ensure that leakage following a postulated design basis accident will not result in doses to the public in excess of 10 CFR Part 100 guidelines and that the emergency core cooling systems will be able to perform their safety functions.

(3) Status:

Review of this topic is being performed by the Division of Operating Reactors (DOR). This effort will continue for the life of operating reactors.

(4) References:

1. Regulatory Guide 1.83, Rev. 1, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes"
2. Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes"
3. 10 CFR Part 50, Appendix A, GDC 30 and 32
4. NUREG-0328, "Regulatory Licensing: Status Summary Report" (Pink Book), 3-27

(5) Basis for Deletion (Related TMI Task, USI, or Other SEP Topic):

USI A-3, A-4, A-5, "Westinghouse, Combustion Engineering, and Babcock and Wilcox Steam Generator Tube Integrity" (NUREG-0649)

The definition of this topic and the references cited are covered by USI A-3, A-4, and A-5. The evaluation for USI A-3, A-4, and A-5 is identical to SEP Topic V-8; therefore, this SEP topic has been deleted.

TOPIC: V-9 Reactor Core Isolation Cooling System (BWR)

(1) Definition:

Reactor core isolation cooling (RCIC) has not been classified as a safety system. On GESSAR, for certain small breaks, GE assumed credit for RCIC as a backup for HPCI. The staff required GE to reclassify the RCIC system on the GESSAR 238 standard NSSS as a safety system.

(2) Safety Objective:

To ensure that the RCIC system is qualified as a safety system where credit is assumed in the safety analysis.

(3) Status:

GE has agreed to reclassify RCIC as a safety system on the GESSAR docket.

TOPIC: V-10.A Residual Heat Removal System Heat Exchanger Tube Failures

(1) Definition:

Residual heat removal (RHR) heat exchangers are designed to remove residual and decay heat so that the reactor can be placed in a safe cold shutdown condition and to maintain core cooling following a postulated loss-of-coolant accident. Some light-water reactors (LWRs) have a pressure control system on the cooling water piping system which maintains the pressure of the cooling water higher than the primary coolant pressure in the primary coolant side of the heat exchanger during plant cooldown operations. A leak in the tubes could result in back leakage of coolant water into the primary loop. Pressure in the cooling water side is maintained higher than that in the primary coolant side so that in the event of a tube failure there would be no leakage of radioactive fluids into the environment. Cooling water passing from the cooling water side of the heat exchanger into the primary coolant water could introduce impurities such as chlorides into the primary coolant system.

(2) Safety Objective:

To assure that impurities from the cooling water system are not introduced into the primary coolant in the event of an RHR heat exhanger tube failure.

(3) Status:

Recently there have been several RHR heat exchanger tube failures at operating BWRs. This issue has been defined as a DOR Category B Technical Activity.

TOPIC: V-10.B Residual Heat Removal System Reliability

(1) Definition:

In all current plant designs, the residual heat removal (RHR) system has a lower design pressure than the reactor coolant system (RCS). In most current designs, the system is located outside of containment and is part of the emergency core cooling system. However, it is possible for the RHR system to have different design characteristics. For example, the RHR system might have the same design pressure as the RCS, or be located inside of containment. The functional, isolation, pressure relief, pump protection, and test requirements for the RHR system are of concern in the safety review of reactor plants. Three types of RHR system designs are defined in Branch Position RSB 5-1.

On June 24, 1976, the Regulatory Requirements Review Committee approved a revision of Standard Review Plan, Section 5.4.7 requiring a capability to go from hot to cold shutdown without offsite power and that all components necessary for cooldown from hot shutdown must be designed to safety grade seismic I standards, and be operable from the control room. System must be designed to meet the single failure criterion.

(2) Safety Objective:

To ensure reliable plant shutdown capability using safety-grade equipment.

(3) Status:

Because of vendor concern over the impact of the revision, a review was conducted of three PWR plants, and as a result of this review, the staff is proposing that Branch Position RSB 5-1 be modified but that the functional requirements be retained.

(4) References:

1. Standard Review Plan, Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System"
2. Standard Review Plan, Section 5.4.7
3. Memorandum from E. G. Case, NRC, to L. V. Gossick, July 15, 1976.
4. Summary of meeting September 22, 1976, "Capability To Achieve Cold Shutdown Using Safety Grade Systems and Equipment," C. O. Thomas, Docket No. STN-50-545, October 5, 1976.

TOPIC: V-11.A Requirements for Isolation of High- and Low-Pressure Systems

(1) Definition:

Several systems that have a relatively low design pressure are connected to the reactor coolant pressure boundary. The valves that form the interface between the high- and low-pressure systems must have sufficient redundancy and interlocks to assure that the low-pressure systems are not subjected to coolant pressures that exceed design limits. The problem is complicated since under certain operating modes (for example, shutdown cooling and emergency core cooling system injection), these valves must open to assure adequate reactor safety.

(2) Safety Objective:

To assure that adequate measures are taken to protect low-pressure systems connected to the primary system from being subjected to excessive pressure which could cause failures and in some cases potentially cause a loss-of-coolant accident outside of containment.

(3) Status:

A preliminary review of a representative operating plant of each nuclear steam supply system vendor was undertaken. Each low-pressure system connected to the reactor coolant pressure boundary and penetrating the containment was examined. The investigation of a few potential areas of concern is continuing.

TOPIC: V-11.B Residual Heat Removal System Interlock Requirements

(1) Definition:

The residual heat removal (RHR) system is normally located outside of primary containment. It is an intermediate pressure system (usually 600 psia) and has motor-operated valve (MOV) isolation valves connecting it to the reactor coolant system (RCS). If the RHR system were inadvertently connected to the RCS while the RCS is at pressure, a loss-of-coolant accident (LOCA) could result with a loss of all capability of core reflooding since the coolant inventory could be lost outside of containment. To prevent inadvertent opening of the MOVs while the RCS is at pressure, an "OPEN PERMISSIVE" interlock is provided.

If the operator shuts only one of the isolation valves prior to pressurizing the RCS, there is a single valve RCS pressure boundary.

To ensure that both MOVs are shut during a startup and heatup, an "AUTO-CLOSURE" interlock is provided that closes the MOVs.

(2) Safety Objective:

To ensure that operating reactor plants are adequately protected from overpressurizing the RHR system and potentially causing a LOCA outside of containment.

(3) Status:

Several PWR plants do not have the auto closure feature on the RHR, and at least one does not have the open permissive feature. Plants should be reviewed on a case-by-case basis factoring in (1) ASME Code safety valve setting and capacity, (2) interlocks, (3) closure time of MOVs, and (4) location of RHR.

(4) References:

1. Proposed Branch Technical Position RSB-5-1, "Design Requirements of the Residual Heat Removal System"
2. Regulatory Requirements Review Committee Meeting No. 50, June 24, 1976
3. 10 CFR Part 50, Appendix A, GDC 34
4. Memorandum from J. Angelo to R. C. DeYoung, V. Stello, et al., NRC, Subject: "RP-TR Staff Meeting of February 13, 1974 Regarding the Requirements on Shutdown Cooling Systems," February 28, 1974
5. Letter from R. Boyd, NRC, to C. Eicheldinger, Westinghouse Electric Corporation, November 12, 1975
6. Letter from R. Boyd, NRC, to I. Stuart, General Electric Company, November 12, 1975
7. Letter from R. Minogue, NRC, to J. D. Geier, Illinois Power Company, July 8, 1975

TOPIC: V-12.A Water Purity of BWR Primary Coolant

(1) Definition:

Review the primary water monitoring and reactor water cleanup system capabilities, including the water purity, to determine if the maintenance of the necessary purity levels complies with Regulatory Guide 1.56. Review limits on quality control and defined provisions in the event of demineralizer breakthrough.

(2) Safety Objective:

To assure that the water purity level is acceptably low to minimize the potential for intergranular stress corrosion cracking of austenitic stainless steel piping in the reactor coolant pressure boundary of BWRs, including assuring the implementation of Regulatory Guide 1.56.

(3) Status:

Recommendations for specifying the use of additional conductivity measurements and monitoring at various locations, plus the use of pH and chloride measurements, have been submitted to the Division of Standards Development to initiate a revision of Regulatory Guide 1.56, "Maintenance of Water Purity in Boiling Water Reactors," dated June 1973. To date, a generic review of operating BWRs has not been initiated and the current regulatory guide has been implemented in the Technical Specifications of only a few operating plants.

(4) Reference:

Memorandum from R. E. Heineman, to R. B. Minogue, NRC, Subject: "Request for Revision of Regulatory Guide 1.56," 1973

TOPIC: V-13 Waterhammer

(1) Definition:

Waterhammer events have occurred in light water reactor systems. Waterhammer events increase the probability of pipe breaks and could increase the consequences of certain events such as the loss-of-coolant accident. The types of waterhammer, the vulnerable systems (for example, containment spray, service water, feedwater, and steam), and the safety significance of waterhammer have been identified and defined in a staff report of May 1977.

(2) Safety Objective:

To reduce the probability of waterhammer events that have the potential to lead to pipe ruptures in light-water reactor systems which are needed to mitigate the consequences of accidents or that might increase the consequences of accidents previously analyzed.

(3) Status:

Generic review is under way. On March 10, 1977, an interdivisional Division of Operating Reactors/Division of Systems Safety technical review group was formed to investigate the waterhammer issue and to develop a program for its appropriate consideration in licensing reviews and for operating reactors. Consultant work has been performed by CREARE and Livermore Labs.

(4) References:

1. "Water Hammer in Nuclear Power Plants," NRC Staff Report, June 1, 1977
2. Wallis, G. B., P. H. Rothe, et al., "An Evaluation of PWR Steam Generator Water Hammer" (draft), CREARE Inc., February 1977
3. Sutton, S. B., "An Investigation of Pressure Transient Propagation in Pressurized Water Reactor Feedwater Lines" (preliminary), Lawrence Livermore Laboratory, April 15, 1977
4. Office of Nuclear Reactor Regulation, NRR Technical Activities, Category A, Item 1, "Water Hammer," May 1977

(5) Basis for Deletion (Related TMI Task, USI, or Other SEP Topic):

USI A-1, "Water Hammer" (NUREG-0649)

The references cited in this topic were the precursors of USI A-1. The evaluation required for USI A-1 is identical to SEP Topic V-13; therefore, this SEP topic has been deleted.

TOPIC: VI-1 Organic Materials and Postaccident Chemistry

(1) Definition:

(a) Organic materials
The design basis for selection of paints and other organic materials is not documented for most operating reactors. Therefore, there is a need to review the suitability of paints and other organic materials used inside containment, including the possible interactions of the decomposition products of organic materials with engineered safety features (such as filters).

(b) Postaccident chemistry
Low pH solutions that may be recirculated within containment after a design basis accident (DBA) may accelerate chloride stress corrosion cracking which may lead to equipment failure or loss of containment integrity. Low pH may also increase the volatility of dissolved iodines with a resulting increase in radiological consequences.

(2) Safety Objective:

(a) Organic materials
To assure that organic paints and coatings used inside containment do not behave adversely during accidents when they may be exposed to high radiation fields. In particular, the possibility of coatings clogging sump screens should be minimized.

(b) Postaccident chemistry
To assure that appropriate methods are available to raise or maintain the pH of solutions expected to be recirculated within containment after a DBA.

(3) Status:

No work currently being done on this subject for operating plants.

(4) References:

1. Standard Review Plan, Sections 6.1.2 and 6.1.3
2. Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants"

TOPIC: VI-2.A Pressure-Suppression-Type BWR Containments

(1) Definition:

BWR pressure-suppression-type containments (for example, Mark I containment) are subjected to hydrodynamic loads during the blowdown phase of a loss-of-coolant accident (LOCA). These loads have the potential for damaging the components and structures (wetwell, internal structures, restraints, supports, and connected systems) of the containment. During a relief valve blowdown into the suppression pool, the wetwell (torus) shell and safety/relief valve restraints may be overstressed. The hydrodynamic loads were not explicitly identified and included in the design of the Mark I pressure-suppression containment.

(2) Safety Objective:

To assure that the structural integrity of pressure-suppression pool containments is maintained under hydrodynamic loading conditions. It has been determined that the upward forces during the blowdown phase following a LOCA potentially cause the Mark I torus to be lifted, causing failure of connecting systems and supports and leading to loss of the containment integrity. Structural modifications and/or changes in the mode of operation might be necessary to assure adequate safety margins.

(3) Status:

Mark I containments are currently evaluated in a two-step generic review program: The Short-Term Program (STP), completed May 1977, has focused on the determination of the magnitude and significance of hydrodynamic loads. In the Long-Term Program (LTP), to be completed by late 1978, the design basis loads will be finalized and the capability of the containment to withstand the loads within the original design structural margins will be verified. This verification will be based in part on research results from NRC and industry sponsored programs. As a result of the STP, the [illegible] staff required that Mark I plants be operated with a drywell to wetwell differential pressure of at least 1 psi to reduce the vertical loads. In addition, some licensees have modified the torus support system for additional safety margin.

(4) References:

1. NUREG-0328, "Regulatory Licensing: Status Summary Report," (Pink Book) - Generic Issues (April 1977)
   a. Mark I Containment - STP Technical Specifications
   b. Mark I Containment Evaluation - STP
   c. Mark I Containment Evaluation - LTP
   d. Mark I Safety/Relief Valve Line Restraints in Torus
2. Division of Operating Reactors, DOR Technical Activities, Category A, April 1977
   a. Item 2, "Mark I Containment STP"
   b. Item 3, "Mark I Containment LTP"
   c. Item 23, "Mark II Containment"
3. Division of Operating Reactors, DOR Technical Activities, Category B, Item 12, "Assessment of Column Buckling Criteria," May 1977
4. Division of Systems Safety, DSS Technical Activities, Category A, Item 31, "Determination of LOCA and SRV Pool Dynamic Loads for Water Suppression Containments," April 1977

(5) Basis for Deletion (Related TMI Task, USI, or Other SEP Topic):

USI A-7, "Mark I Containment Long-Term Program" (NUREG-0649)

Under this task, a short-term program that evaluated Mark I containment has provided assurance that the Mark I containment system of each operating BWR facility would maintain its integrity and functional capability during a postulated LOCA. A longer term program for BWR facilities, not yet licensed, is planned wherein the NRC staff will evaluate the loads, load combinations, and associated structural acceptance criteria proposed by the Mark I Owners Group prior to the performance of plant-unique structural evaluations. The Mark I Owners Group has initiated a comprehensive testing and evaluation program to define design basis loads for the Mark I containment system and to establish structural acceptance criteria which will assure margins of safety for the containment system which are equivalent to that which is currently specified in the ASME Boiler and Pressure Vessel Code. Also included in their program is an evaluation of the need for structural modifications and/or load-mitigation devices to assure adequate Mark I containment system structural safety margins.

The long-term program for USI A-7 will assure that all plants with Mark I containments are able to tolerate, without loss of function, the LOCA-induced hydrodynamic loads.

The evaluation required by USI A-7 is identical to SEP Topic VI-2.A; therefore, this SEP topic has been deleted.

TOPIC: VI-2.B Subcompartment Analysis

(1) Definition:

The rupture of a high energy line inside a containment subcompartment can cause a pressure differential across the walls of the subcompartment. In the case of a rupture of a PWR main coolant pipe adjacent to the reactor vessel, the subcooled blowdown produces pressure differentials in the annulus between the reactor vessel and the shield wall and also within the reactor vessel across the core barrel. This asymmetric pressure distribution generates loads on the reactor vessel support and on reactor vessel internals, on other equipment supports, and on subcompartment structures which have not been analyzed previously for most operating reactors.

(2) Safety Objective:

To assure that the reactor vessel supports, reactor vessel internals, and other equipment supports and subcompartment structures are designed with an adequate margin against failure due to these loads. The failure could result in a loss of emergency core cooling system capability.

(3) Status:

The staff is reviewing the nuclear steam supply system vendor and architect-engineer design codes used to calculate the loads produced by the asymmetric pressure distribution. Analyses have been completed for a limited number of operating plants. The W TMD code is approved. Bechtel, Gilbert, and United Engineering have submitted codes for review.

(4) References:

1. NUREG-0328, "Regulatory Licensing: Status Summary Report," (Pink Book) - Generic Issue, Item 3-5, "Asymmetric LOCA Loads - PWR," April 1977
2. Division of Operating Reactors, DOR Technical Activities, Category A, Item 32, "Asymmetric LOCA Loads (Reactor Vessel Support Problem)," April 1977
3. Division of Systems Safety, DSS Technical Activities, Category A, Item 14, "Asymmetric Blowdown Loads on Reactor Vessel," April 1977
4. Division of Project Management, DPM Technical Activities, Category A, Item 2, "Reactor Vessel Supports (Asymmetric LOCA Loads From Sudden Subcooled Blowdown)," April 1977

(5) Basis for Deletion (Related TMI Task, USI, or Other SEP Topic):

USI A-2, "Asymmetric Blowdown Loads on Reactor Primary Coolant System" (NUREG-0649)

The references cited in this topic were the precursors of USI A-2. The evaluation required for USI A-2 is identical to SEP Topic VI-2.B (see also SEP Topic III-8.D); therefore, this SEP topic has been deleted.

TOPIC: VI-2.C Ice Condenser Containment

(1) Definition:

Operating experience from the D. C. Cook plant has indicated that sublimation and melting of ice causes a loss of ice inventory and related functional performance problems for the ice consenser system.

(2) Safety Objective:

To assure that a sufficient ice inventory is maintained and to assure the functional performance of the ice condenser system.

(3) Status:

The results of the surveillance program for ice inventory and of the functional performance testing (for example, operation of vent doors) are periodically reviewed by the staff to determine whether the surveillance frequencies should be increased or other action should be taken. Recent surveillance testing indicates that the ice inventory is acceptable and that the D. C. Cook plant can be operated safely for the current fuel cycle. CONTEMPT-4 long-term ice condenser code is expected to be completed by Edgerton, Germeshausen & Grier in October 1977.

(4) Reference:

Division of Operating Reactors, DOR Technical Activities, Category B, Item 53, "Ice Condenser Containments," May 1977

TOPIC: VI-2.D Mass and Energy Release for Postulated Pipe Breaks Inside Containment

(1) Definition:

Review the methods and assumptions of the mass and energy release model, including containment temperatures and pressure response, that were used in previously performed analyses of high-energy line breaks inside containment, including the main steam line break.

(2) Safety Objective:

To assure that design basis conditions (for example, design pressure and temperature) for the containment structure and safety-related equipment are adequate. Determine if the models used in the earlier analyses provide adequate margins of safety when compared with the assumptions and models for current analytical techniques.

(3) Status:

Mass and energy release models, including containment response models, are being reassessed to determine the degree of conservatism in the prediction of the containment pressure and temperature transient resulting from a PWR main steam line break. Application of those models to operating plants is contingent on the results of this reassessment. Mass and energy release models for operating BWR plants are considered in the Mark I Long-Term Program and other BWR review efforts.

(4) References:

1. Division of Operating Reactors, DOR Technical Activities, Category B, May 1977.

a. Item 1, "Pipe Break Inside Containment"
b. Item 2, "Mass and Energy Release to Containment"
2. Division of Systems Safety, DSS Technical Activities, Category A, April 1977
a. Item 7, "Pipe Rupture Design Criteria"
b. Item 29, "Main Steam Line Break Inside Containment"
3. Division of Systems Safety, DSS Technical Activities Report, Item I-C.B.1, "Mass and Energy Release to Containment," December 1975

TOPIC: VI-3 Containment Pressure and Heat Removal Capability

(1) Definition:

The temperature and pressure conditions inside containment due to a postulated loss-of-coolant accident (LOCA), main steam line or feedwater line break depend on the effectiveness of passive heat sinks and active heat removal systems (for example, containment spray system).

(2) Safety Objective:

To assure that the maximum temperature and pressure following a LOCA, main steam, or feedwater line break have been calculated with conservative assumptions and to assure that the passive heat sinks and active heat removal systems provide the full heat removal capability required to maintain the pressure and temperature below the design pressure and temperature of the containment, of safety-related equipment, and instrumentation inside containment.

(3) Status:

The modified CONTEMPT computer code properly accounts for the condensation of superheated steam on containment passive heat sinks. The effects on the design temperatures within the containment are being studied for plants under licensing review.

(4) References:

1. Standard Review Plan, Section 6.2.1.1.A
2. Division of Systems Safety, DSS Technical Safety Activities Report, December 1975
3. Division of Operating Reactors, DOR Technical Activities, Category B, Item 62, "Effective Operation of Containment Sprays in LOCA," May 1977

TOPIC: VI-4 Containment Isolation System

(1) Definition:

Isolation provisions of fluid system of nuclear power plants limit the release of fission products from the containment for postulated pipe breaks inside containment and thus prevent the uncontrolled release of primary system coolant as a result of postulated pipe breaks outside containment. This must be accomplished without endangering the performance of postaccident safety systems. Review the primary containment isolation provisions, in particular, the containment sump lines and fluid systems penetrating containment. Review the design bases for containment ventilation system isolation valves to determine potential releases from the containment. Review the containment purge mode during normal operation with respect to various accident scenarios and consequences including operation of containment purge valves, closure times, and leak tightness.

(2) Safety Objective:

To assure that the primary containment isolation provisions meet the requirements of 10 CFR 50, Appendix A, General Design Criteria 54 through 57. Some of the operating plants may have too few or too many isolation provisions. Containment purging during normal operation in PWRs has raised a concern regarding the ability of the ventilation system isolation valves to close upon receipt of an accident signal. The use of resilient sealing materials in conjunction with the cycling of these valves has resulted in an increased degradation in the leakage integrity of the valve seats. To assure the adequacy of the maintenance and repair schedule to maintain the leakage integrity of the valves for the service life of the plant. To assure that containment purge operations will not adversely affect the consequences of postulated accidents.

(3) Status:

The functional performance of the sump lines and emergency core cooling systems is being reviewed in conjunction with the Appendix K submittals. Implementation criteria are being developed to apply the requirements of Branch Technical Position CSB 6-4 to containment purging practices and to improve the leakage integrity of ventilation system isolation valves.

(4) References:

1. 10 CFR Part 50, Appendix A, GDC 54 through 57
2. Standard Review Plan, Section 6.4.2
3. Standard Review Plan, Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operations"

TOPIC: VI-5 Combustible Gas Control

(1) Definition:

Review the combustible gas control system to determine the capability of the system to monitor the combustible gas concentration in the containment, to mix combustible gases within the containment atmosphere, and to maintain combustible gas concentrations below the combustion limits (for example, by recombination, dilution, or purging). For facilities which share recombiners (portable) between units or sites, determine that the recombiners can be made available within a suitable time. For facilities which utilize purging as a primary means of combustible gas control, determine the radiological consequences of the system operation. Reevaluate hydrogen production and accumulation analysis to consider (1) reduction of Zr/water reaction on the basis of five times the Appendix K calculation amount and (2) potential increases in hydrogen production from corrosion of metals inside containment.

(2) Safety Objective:

To prevent the formation of combustible gas explosive concentrations in the containment or in localized regions within containment, following a postulated accident; to assure that the radiological consequences of the system operation are acceptable.

(3) Status:

Proposed 10 CFR 50.44 would permit a BWR licensee to propose an alternate combustible gas control system in lieu of inerting. Four such proposals for containment atmosphere dilution systems are currently under review, and the COGAP II computer code is being revised to perform the system evaluations.

(4) References:

1. Proposed rule 10 CFR Part 50, Section 50.44
2. Division of Operating Reactors, DOR Technical Activities, Category A, Item 8, "Containment Purge During Normal Operation," April 1977
3. Division of Operating Reactors, DOR Technical Activities, Category A, Item 14, "Inerting Requirements/CAD," April 1977
4. Standard Review Plan, Branch Technical Position CSB 6-2, "Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident"
5. Standard Review Plan, Section 6.2.5

(5) Basis for Deletion (Related TMI TASK, USI, or Other SEP Topic):

(a) TMI Action Plan Task II.B.7, "Analysis of Hydrogen Control" (NUREG-0660)

As a result of TMI Task II.B.7, short- and long-term rulemaking to amend 10 CFR 50.44 has been initiated. The short-term rulemaking (interim rule) requires that all Mark I and Mark II containments be inerted. It also requires that the owners of all plants with other containments perform certain analyses of accident scenarios involving hydrogen releases and furnish the staff with a proposed approach for mitigating these hydrogen releases.

The longer-term rulemaking will address both degraded core and melted core issues. In the area of hydrogen control, it will prescribe requirements that are appropriate for operating plants as well as for plants under construction.

(b) USI A-48, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment" (NUREG-0705)

Under USI A-48, a Task Action Plan has been defined and is being developed that encompasses the concerns in the Definition and the Safety Objective of SEP Topic VI-5.

The evaluation required by TMI II.B.7 and USI A-48 is identical to SEP Topic VI-5; therefore, this SEP topic has been deleted.

TOPIC: VI-6 Containment Leak Testing

(1) Definition:

Certain requirements of primary reactor containment leakage testing for water-cooled power reactors as described in Appendix J to 10 CFR Part 50 (issued February 1973) have been found to be conflicting, impractical for implementation, or subject to a variety of interpretations. Review the primary reactor containment leak testing program for operating nuclear plants.

(2) Safety Objective:

To assure that the containment leak testing program provides a conservative assessment of the leakage rate through individual leakage barriers and to assure that proper maintenance and repairs are conducted during the service life of the containment. The testing acceptance criteria are established to ensure that containment leakage following a postulated accident will not result in offsite doses exceeding 10 CFR 100 guidelines.

(3) Status:

A generic review for compliance with Appendix J and the review of requested exemptions to the regulation is currently underway. Proposed revisions to Appendix J to improve the testing requirements are under development.

(4) References:

1. 10 CFR Part 50, Appendix J
2. 10 CFR Part 50, Appendix A, GDC 52 and 53
3. NUREG-0328, "Regulatory Licensing: Status Summary Report" (Pink Book), Generic Issue 3-10, "Containment Leak Testing - Appendix J," April 1977
4. Division of Operating Reactors, DOR Technical Activities, Category B, Item 33, "Containment Leak Testing Requirements," May 1977
5. Division of Systems Safety, DSS Technical Activities, Category A, Item 30, "Containment Leak Testing," April 1977

TOPIC: VI-7.A.1 Emergency Core Cooling System Reevaluation To Account for Increased Reactor Vessel Upper Head Temperature

(1) Definition:

Loss-of-coolant accident (LOCA) analyses for all Westinghouse reactors were conducted assuming that the water in the upper head region of the reactor vessel was the same as the inlet water temperature because of a bypass flow from the downcomer to the upper head. Temperature measurements made by Westinghouse indicate that the actual temperature of the upper head fluid exceeds cold leg temperature by 50 to 75% of the difference between hot leg and cold leg (inlet) temperature. All operating reactors were required to resubmit LOCA analyses using hot leg temperature for the upper head volume.

(2) Safety Objective:

To provide revised LOCA analyses with correct upper head temperatures to assure that peak clad temperature limits are not exceeded.

(3) Status:

Revised analyses have been received from all Westinghouse plants. All but three have been reviewed and approved.

TOPIC: VI-7.A.2 Upper Plenum Injection

(1) Definition:

Emergency core cooling system (ECCS) evaluation of Westinghouse two-loop plants was performed assuming that low pressure pumped injection is delivered directly to the lower plenum. However, ECC coolant is delivered directly into the upper plenum. Interaction of the cold injection water with the steam exiting from the core during refill and reflood and the heat transfer effects during the downward passage to the lower plenum have not been adequately considered.

(2) Safety Objective:

To provide assurance that existing analyses with Westinghouse two-loop plants are acceptable either by showing that the present analyses are conservative, or by developing a new ECCS model which considers upper plenum injection.

(3) Status:

The staff met with the licensees and Westinghouse on January 11 and 26, 1977. The staff requested that the licensees formally submit the information presented at the January 26, 1977 meeting. Two Westinghouse reports have been received to date. The staff is continuing to evaluate the problem. Research requested by the Office of Nuclear Reactor Regulation and performed by the Office of Nuclear Regulatory Research in the semiscale facility provided basis for evaluation.

TOPIC: VI-7.A.3 Emergency Core Cooling System Actuation System

(1) Definition:

Review the emergency core cooling system (ECCS) actuation system with respect to the testability of operability and performance of individual active components of the system and of the entire system as a whole under conditions as close to the design condition as practical.

(2) Safety Objective:

To assure that all ECCS components (for example, valves and pumps) are included in the component and system test. To assure that the frequency and scope of the periodic testing are adequate and meet the requirements of General Design Criterion 37.

(3) Status:

New applications (construction permit and operating license) are reviewed in accordance with the Standard Review Plan and the references listed below. No specific activity for operating reactors is in progress.

(4) References:

1. Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Function"
2. Standard Review Plan, Branch Technical Position EICSB-25, "Guidance for the Interpretation of General Design Criterion 37 for Testing the Operability of the Emergency Core Cooling System as a Whole"
3. 10 CFR Part 50, Appendix A, GDC 37

TOPIC: VI-7.A.4 Core Spray Nozzle Effectiveness

(1) Definition:

Core spray systems are designed with a nozzle or a set of nozzles arranged above the core in such a way that, following a LOCA, a spray of water will be distributed over the top of the core so that each fuel bundle will receive a specified minimum flow which will provide adequate core cooling. Recent test data for a single nozzle in a steam environment noted partial or complete collapse of the spray cone and/or a shift in the direction of spray. These effects were not included in earlier full scale spray tests in air.

(2) Safety Objective:

To assure adequate spray cooling following a LOCA.

(3) Status:

The NRC has reviewed and accepted spray system performance for multiple nozzle spray systems, but has not accepted spray systems with a single overhead spray nozzle. Recent tests in Florida on the Big Rock Point spray nozzle indicate incomplete core coverage. As a result of these tests, NRC is requesting further testing by GE of multiple spray nozzles.

(4) References:

1. Letter from K. Goller, NRC, to operating reactor branch chiefs, Subject: "Generic Issue - Effects of Steam Environment on Core Spray Distribution for Non-jet Pump BWRs," December 7, 1976
2. General Electric, GE Topical Report NEDO-10846, "BWR Core Spray Distribution"

TOPIC: VI-7.B Engineered Safety Feature Switchover From Injection to Recirculation Mode (Automatic Emergency Core Cooling System Realignment)

(1) Definition:

Most PWRs require operator action to realign emergency core cooling (ECC) systems for the recirculation mode following a LOCA.

We have been requiring, on an ad hoc basis, some automatic features to realign the ECCS from the injection to the recirculation mode of operation.

(2) Safety Objective:

To increase the reliability of long-term core cooling by not requiring operator action to change system realignment to the recirculation mode.

(3) Status:

A draft Branch Technical Position has been prepared which covers both ECC and containment spray systems. The proposed position is awaiting review by the Regulatory Requirements Review Committee.

(4) Reference:

American National Standards Institute, Draft ANSI Standard N 660, "Proposed American National Standard Criteria for Safety-Related Operator Actions"

TOPIC: VI-7.C Emergency Core Cooling System (ECCS) Single-Failure Criterion and Requirements for Locking Out Power to Valves, Including Independence of Interlocks on ECCS Valves

(1) Definition:

The physical locking out of electrical sources to specific motor-operated valves required for the engineered safety functions of ECCS has been required, based on the assumption that a spurious electrical signal at an inopportune time could activate the valves to the adverse position; for example, closed rather than open, or opened rather than closed. There is some concern that interlock circuitry on ECCS valves may not be independent such that a single failure of an interlock due to equipment malfunction or operator error could defeat more than one interlock and cause the valves to be cycled to the wrong position.

(2) Safety Objective:

To ensure that all power-operated valves which could affect emergency core cooling (ECC) system performance by being in the wrong position have power removed except when in use. This will ensure that ECC systems are not defeated by having a valve in the wrong position.

(3) Status:

The staff plans to reconsider EICSB BTP-18 and RSB BTP-6-1.

TOPIC: VI-7.C.1 Appendix K--Electrical Instrumentation and Control Re-reviews

(1) Definition:

During the Appendix K reviews of some facilities initially considered, a detailed electrical instrumentation and control review was not performed. Re-review the modified ECCS of these facilities to confirm that it is designed to meet the most limiting single failure.

(2) Safety Objective:

To assure that the modified ECCS is designed to meet the most limiting (design basis) single failure.

(3) Status:

No current activity in the Division of Operating Reactors.

(4) References:

1. Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems"
2. Institute of Electrical and Electronics Engineers, IEEE Std. 308, "Standard Criteria for Class IE Electric Systems for Nuclear Power Generating Stations"

TOPIC: VI-7.C.2 Failure Mode Analysis (Emergency Core Cooling System)

(1) Definition:

Failure modes and effects criticality analyses (FMECA) would be conducted for the purpose of systematically determining potential single failures in emergency core cooling (ECC) systems.

(2) Safety Objective:

To determine if single failures exist in ECC system as an aid in assessing overall plant safety.

(3) Status:

FMECAs have been conducted on the hydraulic portion of ECC systems of representative plant types. In addition, single-failure analyses were performed on each plant as a part of the required Appendix K analysis except for those plants with stainless steel clad cores.

TOPIC: VI-7.C.3 Effect of PWR Loop Isolation Valve Closure During a Loss-of-Coolant Accident on Emergency Core Cooling System Performance

(1) Definition:

Some PWRs are equipped with loop isolation valves. The effect of spurious closure of a loop isolation valve during a LOCA has never been analyzed. To ensure emergency core cooling system (ECCS) performance, power in some cases has been removed from loop isolation valves to prohibit spurious closure.

(2) Safety Objective:

To assure that all plants with loop isolation valves have power removed during operation, or that other acceptable measures are taken to preclude inadvertent closing.

(3) Status:

In most cases power has been removed from loop isolation valves, and this is confirmed as part of staff ECCS performance evaluations. This has not been confirmed for all plants with loop isolation valves.

TOPIC: VI-7.D Long-Term Cooling Passive Failures (for example, Flooding of Redundant Components)

(1) Definition:

The General Design Criteria require that the emergency core cooling systems (ECCSs) shall be capable of providing adequate core cooling following a loss-of-coolant accident, assuming a single failure in emergency core cooling systems. The staff assumes the single failure to be either an active failure during the injection phase, or an active or passive failure during the long-term recirculation phase. The physical layouts of engineered safety feature pumps and components on some pressurized water reactors make them vulnerable to flooding that might result from passive failures in system piping. Protection for pipe cracks or ruptures is not required because of the low probability of occurrence during the ECCS recirculation mode.

(2) Safety Objective:

To provide for increased reliability of ECCSs by assuring that passive failures will not cause flooding and failure of ECCS valves and equipment.

(3) Status:

Issue identified by Fluegge in letter to Rowden, October 24, 1976. Staff response was prepared which concluded that "...consideration of this issue does not warrant revisions to any existing licenses or changes in present priority for addressing the treatment of passive failures subsequent to a LOCA. ECCS passive failure criteria being implemented by the staff require considerations of additional leakage but not pipe breaks beyond the initiating LOCA."

(4) Reference:

NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum From Director, NRR, to NRR Staff," Issue No. 7, "Passive Failures Following a Loss-of-Coolant Accident," December 1976

TOPIC: VI-7.E Emergency Core Cooling System Sump Design and Test for Recirculation Mode Effectiveness

(1) Definition:

Following a loss-of-coolant accident in a PWR, an emergency core cooling system (ECCS) automatically injects water into the system to maintain core cooling. Initially, water is drawn from a large supply tank. Water discharging from the break and containment spray collects in the containment building sump. When the supply tank has emptied to a predetermined level, the ECCS is switched from the "injection" mode to the "recirculation" mode. Water is then drawn from the containment building sump.

ECCSs are required to operate indefinitely in this mode to provide decay heat removal. Certain flow conditions could occur in the sump; which could cause pump failures. These include entrained air, prerotation or vortexing, and losses leading to deficient net positive suction head.

(2) Safety Objective:

To confirm effective operation of ECCSs in the recirculation mode.

(3) Status:

Confirmation through preoperational testing is now required on all construction permits. Staff has been accepting scaled tests in lieu of preoperational tests at the operating-license stage. Some plants have required modification to achieve vortex control.

(4) Reference:

Regulatory Guide 1.79, "Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors," (paragraph b(2))

(5) Basis for Deletion (Related TMI Task, USI, or Other SEP Topic):

USI A-43, "Containment Emergency Sump Reliability" (NUREG-0510 and NUREG-0660)

The definition of this topic and the references cited are covered by USI A-43. The evaluation for USI A-43 is identical to SEP Topic VI-7E; therefore, this SEP topic has been deleted.

TOPIC: VI-7.F Accumulator Isolation Valves Power and Control System Design

(1) Definition:

For many loss-of-coolant accidents, the performance of the ECCS in PWR plants depends upon the proper functioning of the accumulators. The motor-operated isolation valve, provided between the accumulator and the primary system, must be considered to be "operating bypass" (IEEE 279-1971) because, when closed, it prevents the accumulator from performing the intended protective function. The motor-operated isolation valve should be designed against a single failure that can result in a loss of capability to perform a safety function.

(2) Safety Objective:

To assure that the accumulator isolation valve meets the "operation bypass" requirements of IEEE 279-1971, which states that the bypass of a protective function will be removed automatically whenever permissive conditions are not met. To assure that a single failure in the electrical system or single operator error cannot result in the loss of capability of an accumulator to perform its safety function.

(3) Status:

Staff positions listed below are implemented on new applications. No systematic review program for operating reactors exists.

(4) References:

1. Institute of Electrical and Electronics Engineers, IEEE Std. 279-1971, "Criteria for Protection System for Nuclear Power Generating Stations"
2. Standard Review Plan, Branch Technical Position EICSB-4, "Requirements on Motor-Operated Valves in the ECCS Accumulator Lines"
3. Standard Review Plan, Branch Technical Position EICSB-18, "Application of Single Failure Criteria to Manually-Controlled Electrically Operated Valves"

TOPIC: VI-8 Control Room Habitability

(1) Definition:

Control rooms in operating plants may not fully comply with General Design Criterion 19. This review should include, but not be limited to, analysis of the control room air infiltration rate, ventilation system isolability and filter efficiency, shielding, emergency breathing apparatus, short distance atmospheric dispersion, operator radiation exposure, and onsite toxic gas storage proximity.

(2) Safety Objective:

To assure that the plant operators can safely remain in the control room to manipulate the plant controls after an accident.

(3) Status:

The Division of Operating Reactors now reviews control room habitability in operating plants when related licensing actions (for example, assessment of BWR containment air dilution system post-LOCA radiological impact) require it. The Division of Site Safety and Environmental Analysis has a technical assistance contract with the National Bureau of Standards to measure the control room air infiltration rate at a few operating plants. These measurements will be used to gauge the conservatism of the assumed air infiltration rates currently used by NRC. Some reviews are now in progress for plants we have reason to believe do not meet General Design Criterion 19 (San Onofre Nuclear Generating Station Unit 1, Vermont Yankee, St. Lucie).

(4) References:

1. Standard Review Plan, Section 6.4
2. 10 CFR Part 50, Appendix A, GDC 19
3. Murphy, K. G., and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," in Proceedings of the Thirteenth AEC Air Cleaning Conference, August 1974
4. Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release"
5. Regulatory Guide 1.95, Rev. 1, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release"

(5) Basis for Deletion (Related TMI Task, USI, or Other SEP Topic):

- TMI Action Plan Task III.D.3.4, "Control Room Habitability Requirements" (NUREG-0737)

  The review criteria required by Task III.D.3.4 (NUREG-0737, pp. 3-197) are identical to the review criteria specified in the Definition and References of SEP Topic VI-8; therefore, this SEP topic has been deleted.

TOPIC: VI-9 Main Steam Line Isolation Seal System (BWR)

(1) Definition:

Operating experience has indicated that there is a relatively high failure rate and variety of failure modes for components of the main steam isolation valve leakage control system in certain operating BWRs.

(2) Safety Objective:

To assure that leakage rate limits are not exceeded and the resulting calculated offsite doses do not exceed 10 CFR Part 100 guidelines using the staff's assumptions.

(3) Status:

Experience from surveillance testing as reported in recent licensee event reports is compiled by the Division of Operating Reactors to serve as a basis for identifying design improvements and for preparing recommendations for future revisions to Regulatory Guide 1.96.

(4) References:

1. Division of Operating Reactors, DOR Technical Activities, Category B, "Main Steam Line Leakage Control System," May 1977
2. Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants"
3. Standard Review Plan, Section 6.7

TOPIC: VI-10.A Testing of Reactor Trip System and Engineered Safety Features, Including Response-Time Testing

(1) Definition:

Review the reactor trip system (RTS) and engineered safety features (ESF) test program to verify RTS and ESF operability on a periodic basis and to verify RTS and ESF response time.

(2) Safety Objective:

To assure the operability of the RTS and ESF, on a periodic basis, including verification of sensor response times. To ensure that the RTS and ESF test program demonstrates a high degree of availability of the systems and the response times assumed in the accident analyses are within the design specifications.

(3) Status:

The test program of the RTS and ESF of new license applications is reviewed in accordance with the Standard Review Plan, including applicable Branch Technical Positions. Some licensees have agreed to perform response-time measurements. Operability testing is probably performed, in one form or another, for most licensees of operating reactors.

(4) References:

1. Standard Review Plan, Branch Technical Position EICSB-24, "Testing of Reactor Trip System and Engineered Safety Feature Actuation System Sensor Response Times"
2. Memorandum from V. Stello, NRC, to V. A. Moore, Subject: "GESSAR Second Round of Questions No. 2 and No. 9," October 12, 1973
3. Regulatory Guides
   1.22, "Periodic Testing of Protection System Actuation Functions"
   1.105, "Instrument Setpoints"
   1.118, "Periodic Testing of Electric Power and Protection Systems"

TOPIC: VI-10.B Shared Engineered Safety Features, Onsite Emergency Power, and Service Systems for Multiple Unit Stations

(1) Definition:

The sharing of engineered safety features (ESF) systems, including onsite emergency power systems, and service systems for a multiple-unit facility can result in a reduction of the number and of the capacity of onsite systems to below that which normally is provided for the same number of units located at separate sites. Review these shared systems for multiple-unit stations.

(2) Safety Objective:

To assure that: (1) the interconnection of ESF, onsite emergency power, and service systems between different units is not such that a failure, maintenance, or testing operation in one unit will affect the accomplishment of the protection function of the systems(s) in other units; (2) the required coordination between unit operators can cope with an incident in one unit and safe shutdown of the remaining units(s); and (3) system overload conditions will not arise as a consequence of an accident in one unit coincident with a spurious accident signal or any other single failure in another unit.

(3) Status:

A systematic review of shared ESF, onsite emergency power, and service systems for operating multiple-unit stations is not being conducted. The EICSB Branch Technical Position is applied in the review of new licensee applications.

(4) References:

1. Standard Review Plan, Branch Technical Position EICSB-7, "Shared Onsite Emergency Electric Power Systems for Multi-Unit Stations"
2. Regulatory Guide 1.81, "Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants"

TOPIC: VII-1.A Isolation of Reactor Protection System From Nonsafety Systems, Including Qualification of Isolation Devices

(1) Definition:

Nonsafety systems generally receive control signals from the reactor protection system (RPS) sensor current loops. The nonsafety sensor circuits are required to have isolation devices to ensure the independence of the RPS channels. Requirements for the design and qualification of isolation devices are quite specific. Recent operating experience has shown that some of the earlier isolation devices or arrangements at operating plants may not be effective.

(2) Safety Objective:

To verify that operating reactors have RPS designs which provide effective and qualified isolation of nonsafety systems from safety systems to assure that safety systems will function as required.

(3) Status:

A limited generic review of isolation devices is being performed by the Division of Operating Reactors as part of a followup on LER No. 76-42/IT for Calvert Cliffs Unit 1 (TAC 6696). This limited generic review should be complete by August 1, 1977.

(4) References:

1. Licensee Event Report No. 76-42/IT, Calvert Cliffs Unit 1 (Technical Assignment Control (TAC) No. 6696)
2. Standard Review Plan, Section 7.2

TOPIC: VII-1.B Trip Uncertainty and Setpoint Analysis Review of Operating Data Base

(1) Definition:

As a result of Issue No. 13 in NUREG-0138 (Ref. 1) the staff is conducting a survey of plants at the operating-license stage of review to more specifically identify the margin between actual allowable trip parameter limits (from safety analyses standpoint) and actual reactor protection system (RPS) setpoints specified in the Technical Specifications. To clearly identify the setpoint margins, both the ultimate allowable and the specified nominal setting will be identified in the Technical Specifications.

(2) Safety Objective:

To assure that the margins between the allowable trip parameters and the actual RPS setpoints are adequate and properly identified.

(3) Status:

Implementation letters have been sent to the current applicants for operating licenses. The Technical Specifications for operating reactors are only being changed to include both values if a particular plant is converting to Standard Technical Specifications.

(4) References:

1. NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum From Director, NRR, to NRR Staff," Issue No. 13, "Instrument Trip Setpoints in Standard Technical Specifications," November 1976
2. Memorandum from V. Stello, NRC, to R. Boyd, Subject: "Instrument Trip Setpoint Values," February 18, 1977

3. Division of Operating Reactors, DOR Technical Activities, Category B, Item 29, "Instrument Trip Setpoints on Standard Technical Specifications," May 1977

TOPIC: VII-2 Engineered Safety Features System Control Logic and Design

(1) Definition:

During the staff review of the safety injection system (SIS) reset issue (Ref. 1) the staff determined that the engineered safety features actuation systems (ESFASs) at both PWRs and BWRs may have design features that raise questions about the independence of redundant channels, the interaction of reset features and individual equipment controls, and the interaction of the ESFAS logic that controls transfers between onsite and offsite power sources. Review the as-built logic diagrams and schematics, operator action required to supplement the ESFAS automatic actions, the startup and surveillance testing procedures for demonstrating ESFAS performance.

Several specific concerns exist with regard to the manual SIS reset feature following a LOCA: (1) If a loss of offsite power occurs after reset, operator action would be required to remove normal shutdown cooling loads from the emergency bus and reestablish emergency cooling loads. Time would be critical if the loss of offsite power occurred within a few minutes following a LOCA. (2) If loss of offsite power occurs after reset, some plants may not restart some essential loads such as diesel cooling water. (3) The plant may suffer a loss of ECCS delivery for some time period before emergency power picks up the ECCS system.

Review the ESF system control logic and design, including bypasses, reset features, and interactions with transfers between onsite and offsite power sources.

(2) Safety Objective:

To assure that the ESFASs are designed and installed so that the necessary automatic control of engineered safety features equipment can be accomplished when required.

(3) Status:

A review of ESFASs of operating PWRs is being performed by the Division of Operating Reactors as part of the followup action to Reference 1 (to be completed end of 1977).

(4) References:

1. NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum From Director, NRR, to NRR Staff," Issue No. 4, "Loss of Offsite Power Subsequent to Manual Safety Injection Reset Following a LOCA," November 1976
2. Division of Operating Reactors, DOR Technical Activities Category A, Item 22, "Loss of Offsite Power Subsequent to Manual Reset," April 1972

3. Regulatory Guide 1.41, "Preoperational Testing of Redundant Onsite Electric Power Systems To Verify Proper Load Group Assignments"

TOPIC: VII-3 Systems Required for Safe Shutdown

(1) Definition:

Review plant systems that are needed to achieve and maintain a safe shutdown condition of the plant, including the capability for prompt hot shutdown of the reactor from outside the control room. Included also, a review of the design capability and method of bringing a PWR from a high-pressure condition to low-pressure cooling assuming the use of only safety-grade equipment.

(2) Safety Objective:

(1) To assure the design adequacy of the safe shutdown system to (i) initiate automatically the operation of appropriate systems, including the reactivity control systems, such that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences or postulated accidents and (ii) initiate the operation of systems and components required to bring the plant to a safe shutdown.

(2) To assure that the required systems and equipment, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown are located at appropriate locations outside the control room and have a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

(3) To assure that only safety-grade equipment is required for a PWR plant to bring the reactor coolant system from a high-pressure condition to a low-pressure cooling condition.

(3) Status:

A survey of remote shutdown capability of operating plants was performed some time ago by the Division of Operating Reactors. A technical activity has been proposed by the Division of Project Management (see reference below) regarding safety objective (3). No other activities are in progress.

(4) Reference:

Division of Project Management, DPM Technical Activities, Category A, Item 7, "Isolating Low Pressure Systems Connected to the RCPB," April 1977

TOPIC: VII-4 Effects of Failure in Nonsafety-Related Systems on Selected Engineered Safety Features

(1) Definition:

Potential combinations of transients and accidents with failures of nonsafety-related control systems were not specifically evaluated in the original safety analysis of currently operating reactor plants. Review the effects of control system malfunctions as initiating events for anticipated transients and also as failures concurrent with or subsequent to anticipated events or postulated accidents initiated by a different malfunction (for example, the effect of the loss of the plant air system on the plant control and monitoring system). A complete discussion is provided in Reference 1.

(2) Safety Objective:

To assure that any credible combination of a nonsafety-related system failure with a postulated transient or accident will not cause unacceptable consequences.

(3) Status:

A technical assistance contract with Oak Ridge National Laboratory for failure mode analyses of control systems was initiated to determine sensitive areas of the plant designs. The results of this program in conjunction with the results of the failure mode and effects analyses for transients and accidents being performed under contract by Idaho Nuclear Engineering Laboratory should provide a basis for any new review and safety requirements.

(4) References:

1. NUREG-0153, "Staff Discussion of Twelve Additional Technical Issues Raised by Responses to November 3, 1976 Memorandum from Director, NRR, to NRR Staff," Issue 22, "Systematic Review of Normal Plant Operation and Control System Failures," December 1976
2. Memorandum from V. Stello, NRC, to R. J. Hart, December 23, 1976, NRR letter No. 46.
3. Division of Operating Reactors, DOR Task Force Report on SEP, Appendix B (TFL 118), November 1976
   a. Item 33, "Safety Related Control Power"
   b. Item 34, "Safety Related Instrumentation Power"
   c. Item 56, "Effect of Failure in Non-Safety Related Systems During Design Basis Events"
   d. Item 57, "Loss of Plant Air System (Effect on Plant Control and Monitoring)"
   e. Item 77, "Safety Related Control and Instrument Power"
4. Directorate of Operational Technology, DOT Recommended List of SEP Subjects, C DOT 102, Item 100z, "Loss of Plant Air System (Effect on Plant Control and Monitoring)," Spring 1977

(5) Basis for Deletion (Related TMI Task, USI, or Other SEP Topic):

(a) USI A-47, "Safety Implications of Control System" (NUREG-0705 and NUREG-0606)

The issue defined in Reference 1 (NUREG-0153, Item 22) is as follows:

> In evaluating plant safety, the effects of control system malfunctions should be reviewed as initiating events for

anticipated transients and also as failures that could occur concurrently subsequent to postulated anticipated events (initiated by a different malfunction) or postulated accidents.

The issue defined in USI A-47 is, in part, as follows:

> This issue concerns the potential for transients or accidents being made more severe as a result of the failure or malfunction of control systems. These failures or malfunctions may occur independently, or as a result of the accident or transient under consideration.

(b) USI A-17, "Systems Interactions in Nuclear Power Plants" (NUREG-0649 and NUREG-0606)

The purpose of this task is to develop a method for conducting a disciplined and systematic review of nuclear power plant systems, for both process function couplings of systems and space couplings, to identify the potential sources and types of systems interactions that are determined to be potentially adverse.

A report has been developed, "Final Report - Phase 1 Systems Interaction Methodology Applications Program," NUREG/CR-1321, SAND 80-0384, whose objectives are:

1. To develop a methodology for conducting a disciplined and systematic review of nuclear power plant systems which facilitates identification and evaluation of systems interactions that affect the likelihood of core damage.

2. To use the methodology to assess the Standard Review Plan to determine the completeness of the plan in identifying and evaluating a limited range of systems interactions.

The work done under USI A-17 may be useful in the development of USI A-47.

The Definition of USI A-47 is identical to that of Topic VII-4; therefore, this SEP topic has been deleted.

TOPIC: VII-5 Instruments for Monitoring Radiation and Process Variables During Accidents

(1) Definition:

The adequacy of the instruments for monitoring radiation and process variables during accidents has not been reviewed for conformance with Regulatory Guide 1.97. A generic review is planned to assess the licensee's existing or proposed monitoring instruments during and following accidents to determine the adequacy of their range, response, and qualifications, and to determine the sufficiency of the variables to be monitored. Certain instruments to monitor conditions beyond the design basis accidents will also be required in accordance with an Regulatory Requirements Review Committee (RRRC) determination (Reference 3).

(2) Safety Objective:

To assure that plant operators and emergency response personnel have available sufficient information on plant conditions and radiological releases to determine appropriate in-plant and offsite actions throughout the course of any accident. The instrumentation should also provide recorded transient or trend information necessary for postaccident evaluation of the event. The ability to follow the course of accidents beyond the design basis accidents is also required.

(3) Status:

Generic review of instrumentation to follow the course of accidents in operating plants and in all plants now under construction or seeking a construction permit will begin with the issuance of Regulatory Guide 1.97, Revision 1, this year. Submittals describing the facilities' postaccident instrumentation will be obtained from all operating licensees and reviewed by the end of 1978. The implementation of Regulatory Guide 1.97, Revision 1 on operating plants is proceeding independent of the SEP. The Regulatory Requirements Review Committee has determined that Revision 1 to Regulatory Guide 1.97 should be treated as a Category 2 item (backfit on operating plants on a case-by-case basis).

(4) References:

1. Memorandum from H. G. Mangelsdorf (ACRS) to L. M. Muntzing (Regulations), August 14, 1973
2. Memorandum from L. M. Muntzing (Regulation) to H. G. Mangelsdorf (ACRS), November 1, 1973
3. Memorandum from R. B. Minogue (SD) to E. G. Case (NRR), Enclosure, Proposed Revision 1 to Regulatory Guide 1.97, April 4, 1977
4. Standard Review Plan, Section 7.5
5. Standard Review Plan, Section 7.6
6. Standard Review Plan, Section 11.5
7. Memorandum from T. A. Ippolito (EICSB) to Emergency Instrumentation Task Force Members, August 12, 1974
8. NUREG-0153, "Staff Discussion of Twelve Additional Technical Issues Raised by Responses to November 3, 1976 Memorandum from Director, NRR, to NRR Staff," Issue 21, "Instruments for Monitoring Both Radiation and Process Variable During Accidents," December 1976
9. Minutes of Regulatory Requirements Review Committee meeting, January 28, 1977

(5) Basis for Deletion (Related TMI Task, USI, or Other SEP Topic):

TMI Action Plan Task II.F, "Instrumentation and Controls" NUREG-0660 and NUREG-0737

There are three subtasks under Task II.F as follows:

(a) II.F.1 - Additional Accident Monitoring Instrumentation

(b) II.F.2 - Identification of and Recovery From Conditions Leading to Inadequate Core Cooling

(c) II.F.3 - Instruments for Monitoring Accident Conditions

Specific positions on the required instrumentation for II.F.1 and II.F.2 are in NUREG-0737 and Regulatory Guide 1.97, Revision 2 (December 1980). Instrumentation need for II.F.3 is also in Regulatory Guide 1.97, Revision 2.

The emphasis of TMI Task II.F is the monitoring of radiation and process variables; guidance for this relies primarily on Regulatory Guide 1.97. This is identical to the review proposed in Topic VII-5; therefore this SEP topic has been deleted.

TOPIC: VII-6 Frequency Decay

(1) Definition:

In an issue of Reference 1 it is stated that the staff should require that a postulated rapid decay of the frequency of the offsite power system be included in the accident analysis and that the result be demonstrated to be acceptable. Alternatively, the reactor coolant pump (RCP) circuit breakers should be designed to protection system criteria and tripped to separate the pump motors from the offsite power system. Rapid decay of the frequency of the offsite power system has the potential for slowing down or breaking the RCP, thereby reducing the coolant flow rates to levels not considered in previous analyses.

(2) Safety Objective:

To assure that the reactor coolant flow rate will not decrease below those assumed for a flywheel coastdown.

(3) Status:

Oak Ridge National Laboratory, under a technical assistance program, is currently reviewing the frequency decay rate and its effects on RCPs. This program should be completed before the end of this year and this issue resolved.

(4) References:

1. NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum From Director, NRR, to NRR Staff," Issue No. 9, "Frequency Decay," November 1976
2. Division of Operating Reactors, DOR Technical Activities, Category B, Item 27, "Frequency Decay," May 1977

TOPIC: VII-7 Acceptability of Swing Bus Design on BWR-4 Plants

(1) Definition:

The swing bus in the original BWR-4 design was used to provide power from either of two redundant electric sources to the low-pressure coolant injection (LPCI) valves by means of an automatic transfer scheme. A single failure in the transfer circuitry could result in paralleling the two redundant electric power sources, thereby degrading their functional capabilities. Review licensee's swing bus automatic transfer circuitry to verify that it is immune to single failures which could lead to paralleling the two electric power sources.

(2) Safety Objective:

To assure that the swing bus design will not propagate an electrical failure between two redundant power sources due to a single failure in the automatic transfer circuit at the BWR-4 swing bus.

(3) Status:

During the course of generic review for compliance with emergency core cooling system criteria 10 CFR 50.46 and Appendix K, some licensees have elected to modify the LPCI system to take credit for a portion of the LPCI flow. These facilities have replaced the swing bus design with a split bus configuration which complies with the requirements of Regulatory Guide 1.6. Not all facilities required a modification of the LPCI to meet the criteria and have retained the swing bus design.

The issue of the swing bus design was identified in Reference 1 and in addition in a letter from the Advisory Committee on Reactor Safeguards (ACRS) dated December 12, 1976.

(4) References:

1. NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum From Director, NRR, to NRR Staff," Issue No. 3, "Acceptability of Swing Bus Design of BWR-4 Plants," November 1976
2. Regulatory Guide 1.6, "Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems"
3. 10 CFR Part 50, Appendix A, GDC 17
4. Institute of Electrical and Electronics Engineers, IEEE Std. 308, "Standard Criteria for Class IE Electric Systems for Nuclear Power Generating Stations"

TOPIC: VIII-1.A Potential Equipment Failures Associated With Degraded Grid Voltage

(1) Definition:

A sustained degradation of the offsite power source voltage could result in the loss of capability of redundant safety loads, their control circuitry, and the associated electrical components required to perform safety functions.

(2) Safety Objective:

To assure that a degradation of the offsite power system will not result in the loss of capability of redundant safety-related equipment and to determine the susceptibility of such equipment to the interaction of onsite and offsite emergency power sources.

(3) Status:

A program plan has been developed which includes a short-term program for the review of the emergency power systems of operating reactors and a long-term program to identify those conditions affecting the offsite power sources which may require that additional safety measures be taken.

(4) References:

1. NUREG-0090-5, "Report to Congress, Abnormal Occurrences at Millstone 2, July-September 1976," March 1977
2. Memorandum from D. G. Eisenhut, NRC, to K. R. Goller, Subject: "Staff Positions (Short-Term Program)," April 20, 1977
3. Letters to licensees, August 12 and 13, 1976
4. Division of Operating Reactors, DOR Technical Activities, Category A, Item 9, "Potential Equipment Failures Associated with a Degraded Off-site Power Source," April 1977

TOPIC: VIII-2 Onsite Emergency Power Systems (Diesel Generator)

(1) Definition:

Diesel generators, which provide emergency standby power for safe reactor shutdown in the event of total loss of offsite power, have experienced a significant number of failures. The failures to date have been attributed to a variety of causes, including failure of the air startup, fuel oil, and combustion air systems. In some instances, the malfunctions were due to lockout. The information available to the control room operator to indicate the operational status of the diesel generator was imprecise and could lead to misinterpretation. This was caused by the sharing of a single annunciator station by alarms that indicate conditions that render a diesel generator unable to respond to an automatic emergency start signal and alarms that only indicate a warning of abnormal, but not disabling, conditions. Another cause was the wording on an annunciator window which did not specifically say that the diesel generator was inoperable (that is, unable at the time to respond to an automatic emergency start signal), when in fact it was inoperable for that purpose. The review includes the qualification, reliability, operation at low loads, lockout, fuel oil, and testing of diesel generators.

(2) Safety Objective:

To assure that the diesel generator meets the availability requirements for providing emergency standby power to the engineered safety features.

(3) Status:

Under a technical assistance request (in preparation), a thorough evaluation of all reported failures, including a comprehensive evaluation of diesel manufacturer and utility procedures for inspection, maintenance, and operation, will be performed. Letters were sent on March 29, 1977 to all the affected licensees requesting additional information about diesel generator status indication in the control room. Our intention is to require that at least one annunciation be provided in the control room which will alarm whenever the diesel generator is unavailable due to any lockout condition.

(4) References:

1. Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants"
2. NUREG-0328, "Regulatory Licensing: Status Summary Report" (Pink Book), Generic Issue 3-11, "Diesel Generator Lockout," April 1977

TOPIC: VIII-3.A Station Battery Capacity Test Requirements

(1) Definition:

Review the Technical Specification, including the test program, with regard to the requirement for periodic surveillance testing of onsite Class IE batteries and the extent to which the test meets Section 5.3.6 of IEEE Std. 308-1971, to determine battery capacity.

(2) Safety Objective:

To assure that the onsite Class IE battery capacity is adequate to supply dc power to all safety-related loads required by the accident analyses and is verified on a periodic basis. This effort is needed to ensure that the test to determine battery capacity includes (1) an acceptance test of battery capacity performed in accordance with Section 4.1 of IEEE Std. 450-1975; (2) a performance discharge test listed in Table 2 of IEEE Std. 308-1971, performed according to Sections 4.2 and 5.4 of IEEE Std. 450-1975; and (3) a battery service test described in Section 5.6 of IEEE Std. 450-1972, to be performed during each refueling operation.

(3) Status:

The review of station battery capacity test requirements is applicable to all operating reactors. There is no ongoing effort on this subject for operating reactors except for those reactors converting to Standard Technical Specifications.

(4) References:

1. Standard Review Plan, Appendix 7-A, Branch Technical Position EICSB 6
2. Institute of Electrical and Electronics Engineers, IEEE Std. 308-1971, 1974, "Standard Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations"

3. Institute of Electrical and Electronics Engineers, IEEE Std. 450-1975, "Recommanded Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations"
4. Memorandum from J. G. Keppler to R. H. Vollmer, NRC, March 20, 1972
5. Memorandum from V. D. Thomas to R. Carlson, January 18, 1972

TOPIC: VIII-3.B DC Power System Bus Voltage Monitoring and Annunciation

(1) Definition:

Review the dc power system battery, battery charger, and bus voltage monitoring and annunciation design with respect to dc power system operability status indication to the operator. This information is needed so that timely corrective measures can be taken in the event of loss of an emergency dc bus.

(2) Safety Objective:

To assure the design adequacy of the dc power system battery and bus voltage monitoring and annunciation schemes such that the operator can (1) prevent the loss of an emergency dc bus or (2) take timely corrective action in the event of loss of an emergency dc bus.

(3) Status:

The review of the dc power system battery and bus voltage monitoring and annunciation adequacy as it relates to the loss of an emergency dc bus is applicable to all operating reactors. This topic is included in the NRR Technical Activity, "Adequacy of Safety Related DC Power Supplies."

(4) Reference:

Standard Review Plan, Section 8.3.2

TOPIC: VIII-4 Electrical Penetrations of Reactor Containment

(1) Definition:

Review the electrical penetration assembly with respect to the capability to maintain containment integrity during short-circuit current conditions and mechanical integrity during the worst expected fault current vs. time conditions resulting from single random failures of circuit overload protection devices.

(2) Safety Objective:

To assure that all electrical penetrations in the containment structure, whether associated with Class IE circuits or non-Class IE circuits, are designed not to fail from electrical faults during a loss-of-coolant accident.

(3) Status:

The subject of electrical cable penetrations was identified in Reference 1 and has been proposed as a Technical Activity Category A item by the Division of Systems Safety (Reference 2). The purpose of that activity is a reevaluation of the penetrations to clarify and augment the design safety margin.

(4) References:

1. NUREG-0153, "Staff Discussion of Twelve Additional Technical Issues Raised by Responses to November 3, 1976 Memorandum From Director, NRR, to NRR Staff," Issue 18, "Electrical Cable Penetration of Reactor Containment," December 1976
2. Division of Systems Safety, DSS Technical Activity, Category A, Item 36, "Electrical Cable Penetrations of Reactor Containment," April 1977
3. Regulatory Guide 1.63, "Electric Pentration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants"
4. Institute of Electrical and Electronics Engineers, IEEE Std. 317-1976, "Standard for Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations"

TOPIC: IX-1 Fuel Storage

(1) Definition

Review the storage facility for new and irradiated fuel, including the cooling capability and seismic classification of the fuel pool cooling system of the spent fuel storage pool. Specifically review the expansion of the onsite spent fuel storage capacity, including the structural response of the fuel storage pool and the racks, the criticality analysis for the increased number of stored fuel assemblies at reduced spacing, and the capability of the spent fuel cooling system to remove the additional heat load.

(2) Safety Objective:

To assure that new and irradiated fuel is stored safely with respect to criticality ($k_{eff} < 0.95$), cooling capability (outlet temperature $< 150°F$), shielding, and structural capability.

(3) Status:

Approximately two-thirds of the operating reactor plants have requested authorization to increase the storage capacity of their fuel storage pool. The applications are reviewed on a case-by-case basis. New or modified storage rack designs are reviewed against current design criteria; however, the existing pool structure is based on original design criteria.

(4) References

1. Division of Operating Reactors, DOR Technical Activities, Category A, Item 27, "Increase in Spent Fuel Storage Capacity," April 1977
2. American National Standards Institute, ANSI-210, "Design Objectives for Spent Fuel Storage Facilities"

TOPIC: IX-2 Overhead Handling Systems (Cranes)

(1) Definition:

Overhead handling systems (cranes) are used to lift heavy objects in the vicinity of PWR and BWR spent fuel storage facilities and inside the reactor building. If a heavy object (for example, a shielded cask) were to drop on the spent fuel or on the reactor core during refueling, there could be a potential for overexposure of plant personnel and for release of radioactivity to the environment. Review the overhead handling system, including sling and other lifting devices, and the potential for the drop of a heavy object on spent fuel, including structural effects.

(2) Safety Objective:

To assess the safety margins, and improve margins where necessary, of the overhead handling systems to assure that the potential for dropping a heavy object on spent fuel is within acceptable limits and that the potential radiation dose to an individual does not exceed the guidelines of 10 CFR Part 100.

(3) Status:

Regulatory Guide 1.104, "Overhead Crane Handling Systems for Nuclear Power Plants," was issued for comment in February 1976 and references various industry standards. New applications (construction permit and operating license) are reviewed in accordance with APCSB Branch Technical Position 9-1 which is identical to Regulatory Guide 1.104.

The review of overhead handling systems of operating reactor facilities is performed on a generic basis and has also been identified as a DOR Technical Activity Category A.

(4) References:

1. Regulatory Guide 1.104, "Overhead Crane Handling Systems for Nuclear Power Plants"
2. Standard Review Plan, Branch Technical Position APCSB 9-1, "Overhead Handling Systems for Nuclear Power Plants"
3. NUREG-0328, "Regulatory Licensing: Status Summary Report" (Pink Book), Generic Issue 3-22, "Fuel Cask Drop Analysis," April 1977
4. Division of Operating Reactors, DOR Technical Activities, Category A, Item 50, "Control of Heavy Loads Over Spent Fuel," April 1977

(5) Basis for Deletion (Related TMI Task, USI or Other SEP Topic):

- USI A-36, "Control of Heavy Loads Near Spent Fuel" (NUREG-0649)

  The review criteria required by USI A-36 (Standard Review Plan, Section 9.1.4, and NUREG-0554) are identical to the review criteria specified in the References of SEP Topic IX-2 (BTP 9-1 and Regulatory Guide 1.104); therefore, this SEP topic has been deleted.

TOPIC: IX-3 Station Service and Cooling Water Systems

(1) Definition:

Review the station service water and cooling water systems that are required for safe shutdown during normal, operational transient, and accident conditions, and for mitigating the consequences of an accident or preventing the occurence of an accident. These include cooling water systems for reactor system components (components cooling water system), reactor shutdown equipment, ventilation equipment, and components of the emergency core cooling system (ECCS). These systems also include the station service water system, the ultimate heat sink, and the interaction of all the above systems.

The review of these systems includes the pumps, heat exchangers, valves and piping, expansion tanks, makeup piping, and points of connection or interfaces with other systems. Emphasis is placed on the cooling systems for safety-related components such as ECCS equipment, ventilation equipment, and reactor shutdown equipment.

The following specific aspects of those systems will be considered in the review:

(a) Physical separation of redundant cooling water systems that are vital to the performance of engineered safety systems components,

(b) Availability of cooling water to primary reactor coolant pumps,

(c) Requirements for makeup water of cooling water systems,

(d) Effect of water overflow from tanks,

(e) Circulating water system barrier failure protection.

(2) Safety Objective:

To assure that the station service and cooling water systems have the capability, with adequate margin, to meet their design objective. To assure, in particular, that

(a) Systems are provided with adequate physical separation such that there are no adverse interactions among those systems under any mode of operation;

(b) Cooling water is provided to the bearings of the primary reactor coolant pumps by two independent essential service water systems for PWR plants to take credit for core cooling by pump coastdown. In addition, it should be demonstrated that the possibility of simultaneous loss of water in both essential service water systems by valve closure is sufficiently small;

(c) Sufficient cooling water inventory has been provided or that adequate provisions for makeup are available;

(d) Tank overflow cannot be released to the environment without monitoring and unless the level of radioactivity is within acceptable limits;

(e) Vital equipment necessary for achieving a controlled and safe shutdown is not flooded due to the failure of the main condenser circulating water system.

(3) Status:

The station service and cooling water systems of applications currently under review are evaluated in accordance with the Standard Review Plan (Sections 9.2.2 and 10.4.5). Some of the specific concerns identified above are under generic review or have been proposed for a technical activity in the Office of Nuclear Reactor Regulation in accordance with the references below.

(4) References:

1. Letter from R. F. Fraley (ACRS) to L. V. Gossick, Subject: "Analysis of Systems Interactions," November 1, 1976
2. Memorandum from B. C. Rusche to L. V. Gossick, ACRS Subcommittee on Systems Interactions, January 1977
3. Division of Project Management, DPM Technical Activities, Category A, Item DPM-15, "Systems Interactions in Nuclear Power Plants," April 1977
4. Memorandum to R. L. Tedesco, NRC, to D. B. Vassallo, Auxiliary Systems Branch 02 on Yellow Creek Nuclear Plant, Item 010.42, (cooling water for RCP), January 31, 1977
5. Division of Systems Safety, DSS Technical Safety Activities Report, "Cooling Water System Makeup Water Requirements (For Safety Systems)," December 1975
6. NUREG-0328, "Regulatory Licensing: Status Summary Report" (Pink Book), Generic Issue 3-20, "Flood of Equipment Important to Safety (Generic)," April 1977
7. Division of Operating Reactors, DOR Technical Activities, Category A, Item 15, "Flood of Equipment Important to Safety," April 1977

TOPIC: IX-4 Boron Addition System (PWR)

(1) Definition:

Review the boron addition system (PWR), in particular with respect to boron precipitation during the long-term cooling mode of operation following a loss-of-coolant accident.

(2) Safety Objective:

To assure that boron precipitation will not impair the operability of valves or components in the boron addition system which could compromise its capability to control core reactivity during the normal, transient, or emergency shutdown conditions or that would result in flow blockage through the core during the long-term core cooling mode following a loss-of-coolant accident.

(3) Status:

Operating PWR reactors, with the exception of the Combustion Engineering reactors, have been reviewed and found to be acceptable in regard to boron precipitation following a loss of coolant. There are still certain outstanding issues that need to be resolved on this issue for Combustion Engineering reactors. In regard to the precipitation of boron in the boron addition system in both BWRs and PWRs, certain older plants may not have been reviewed in sufficient detail to assure that system reliability is adequate.

(4) Reference:

Standard Review Plan, Section 9.3.4

TOPIC: IX-5 Ventilation Systems

(1) Definition:

Review the design and operation of ventilation systems whose function is to maintain a safe environment for plant personnel and engineered safety features equipment. For example, the function of the spent fuel pool area ventilation system is to provide ventilation in the spent fuel pool equipment areas, to permit personnel access, and to control airborne radioactivity in the area during normal operation, anticipated operational transients, and following postulated fuel handling accidents. The function of the engineered safety feature ventilation system is to provide a suitable and controlled environment for engineered safety feature components following certain anticipated transients and design basis accidents.

(2) Safety Objective:

To assure that the ventilation systems have the capability to provide a safe environment, under all modes of operation, for plant personnel (10 CFR Part 20) and for engineered safety features (for example, to assure that the diesel room has redundant outside air intakes and removed from the exhaust discharge).

(3) Status:

The ventilation systems of plants under current review (construction permit and operating license applications) are currently evaluated in accordance with the Standard Review Plan. No specific issues or concerns have been identified for operating reactor plants.

(4) References:

Standard Review Plan, Sections 9.4.1 through 9.4.5

TOPIC: IX-6 Fire Protection

(1) Definition:

Review the fire protection program of operating reactor plants to determine whether improvements are required in accordance with the APCSB Technical Position 9.5-1, Appendix A (Reference 2). The fire protection program encompasses the components, procedures, and personnel utilized in carrying out all activities of fire protection and includes such things as fire prevention, detection, annunciation, control, confinement, suppression, extinguishment, administrative procedures, fire brigade organization, inspection and maintenance, training, quality assurance, and testing. The review includes such items as: (1) the use of insulation inside the containment and (2) the consequences of the inadvertent release of hydrogen into the plant.

(2) Safety Objective:

To assure that, in case of a fire within the plant, the integrity of the engineered safety features is not compromised and that the safe shutdown capability and control of the plant are not lost.

(3) Status:

A generic review of fire protection for operating plants is under way. All licensees were requested by letter (May 11, 1976) to submit an evaluation of their fire protection program for that plant in comparison with the APCSB Technical Position 9.5-1. Subsequently, in September 1976, the licensees were provided with Appendix A to the BTP 9.5-1 which presents acceptable alternatives for operating plants.

(4) References:

1. NUREG-0050, "Recommendations Related to Browns Ferry Fire," February 1976
2. Standard Review Plan, Branch Technical Position APCSB 9.5-1, Appendix A, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976"

3. Regulatory Guide 1.120, "Fire Protection Guidelines for Nuclear Power Plants"
4. NUREG-0328, "Regulatory Licensing: Status Summary Report" (Pink Book), Generic Issue 3-18, "Fire Protection," April 1977
5. Division of Operating Reactors, DOR Technical Activities, Category A, Item 28, "Fire Protection," April 1977
6. Division of Systems Safety, DSS Technical Activities, Category A, Item 32, "Fire Protection," April 1977
7. Letter from R. F. Fraley, ACRS, to L. V. Gossick, Subject: "Analysis of Systems Interactions - Item 6," November 1, 1976

TOPIC: X Auxiliary Feedwater System

(1) Definition:

Review the auxiliary feedwater system, associated instrumentation, and connection between redundant systems. The review includes the aspects of pump drive and power supply diversity (for example, electrical and steam-driven sources), and the water supply sources for the auxiliary feedwater system.

(2) Safety Objective:

To assure that the auxiliary feedwater system can provide an adequate supply of cooling water to the steam generators for decay heat removal in the event of a loss of all main feedwater. Older PWR plants may not meet the requirement for pump drive and power supply diversity.

(3) Status:

Reviews for new license applications are performed in accordance with the Standard Review Plan. This topic is not under active review for operating plants.

(4) References:

1. Standard Review Plan, Section 10.4.9
2. Standard Review Plan, Branch Technical Position APCSB 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for PWR Plants"

(5) Basis for Deletion (Related TMI Task, USI, or Other SEP Topic):

TMI Action Plan Task II.E.1.1, "Auxiliary Feedwater System Evaluation" (NUREG-0660)

The TMI-2 accident and subsequent investigations and studies highlighted the importance of the auxiliary feedwater (AFW) system in the mitigation of severe transients and accidents. Since then, the AFW systems have come under close scrutiny by the NRC and many improvements have been recommended to enhance the reliability of AFW systems for all plants. The scope of the review outlined in the SEP

Topic X definition is identical to the scope of NUREG-0737, "Clarification of TMI Action Plan Requirements," Item II.E.1.1(2), which requires that each PWR plant licensee:

> Perform a deterministic review of the AFW system using the acceptance criteria of Standard Review Plan Section 10.4.9 and associated Branch Technical Position ASB 10-1 as principal guidance.

The review criteria for the evaluations required by Item II.E.1.1(2) are identical to SEP Topic X; therefore, this SEP topic has been deleted.

<u>TOPIC</u>: XI-1 Appendix I

(1) <u>Definition</u>:

A generic review of all operating plants to determine their capability to comply with Appendix I, 10 CFR 50, and to prevent explosions in the gaseous radwaste system is currently underway.

(2) <u>Safety Objective</u>:

To provide assurance that radioactive gaseous effluents from the facility can be kept "as low as reasonably achievable" as defined in Appendix I, 10 CFR Part 50, and to assure adequate control of the mixture of gases in the gaseous radwaste system to prevent explosions.

(3) <u>Status</u>:

A generic review of all operating reactors (ORs) for their capability to conform with Appendix I, 10 CFR Part 50, is currently under way by the Division of Site Safety and Environmental Analysis. Upon the completion of this review, new gaseous and liquid radiological effluent and monitoring Technical Specifications will be issued to all ORs. This will include new Technical Specifications on gaseous radwaste systems which may contain explosive gas mixtures to meet present criteria. The estimated completion date of this review is 1979.

(4) <u>References</u>:

1. 10 CFR Part 20
2. 10 CFR Part 50, Appendix I
3. 10 CFR Part 50, Appendix A
4. 10 CFR Part 50, Appendix A, GDC 60, 61, 63, and 64
5. Standard Review Plan, Section 11.3

(5) <u>Basis for Deletion</u>

Topic XI-1 is being resolved by the following NRR generic topics: (a) A-02, "Appendix I" and (b) B-35, "Confirmation of Appendix I Models." Resolution of these two generic topics will primarily result in Technical Specification changes and may require some minor hardware changes. At present, nothing more than the addition of monitoring instrumentation is foreseen. The implementation of Appendix I will, therefore, not affect the integrated assessment for SEP plants.

In addition, the implementation of Appendix I will result in limiting conditions for operation to assist licensees in keeping the amount of radioactive material released in effluents to unrestricted areas as low as is reasonably achievable. Since licensees are currently restricted in the types and amounts of effluents they can release, implementation of additional restrictions on releases should not impact operation of the plant.

Based on the above, Topic XI-1 has been deleted from the SEP program.

TOPIC: XI-2 Radiological (Effluent and Process) Monitoring Systems

(1) Definition:

Onsite radiological monitoring systems are used to:

(a) Assess the proper functioning of the process and waste treatment systems,

(b) Assure that radioactive releases do not exceed the appropriate guidelines, and

(c) Measure actual releases to evaluate their environmental impact.

There is concern about the adequacy of radiation monitoring systems. A survey of 12 plants has been initiated. The results of this survey will indicate whether this area needs to be reviewed for all operating plants. Re-review would include the monitor's sensitivity, range, location, and calibration techniques.

(2) Safety Objective:

To provide reasonable assurance that the licensee adequately monitors the releases of radioactive materials in liquid and gaseous effluent and that the releases are properly restricted. To provide assurance that the licensee adequately monitors the operation of equipment that contains or may contain radioactive material.

(3) Status:

A technical assistance program has been initiated at Brookhaven National Laboratory with the scope including the above safety objectives.

(4) References:

1. 10 CFR Part 20, Section 20.106
2. 10 CFR Part 50, Section 50.36a
3. 10 CFR Part 50, Appendix A, GDC 60, 61, 63, and 64
4. 10 CFR Part 50, Appendix I
5. Standard Review Plan, Section 11.5

(5) Basis for Deletion

Topic XI-2 is being resolved by the following NRR generic topics: (a) A-02, "Appendix I" and (b) B-67, "Effluent and Process Monitoring Instrumentation." A-02 is discussed in Topic XI-1. Generic item B-67 was subdivided into four subtasks. The staff believes that events since the inception of B-67 have largely addressed the identified concerns or changed its thinking in regard to their safety significance. The description and bases for deletion of each subtask are presented below.

Subtask 1: Monitoring of Radioactive Materials Released in Effluents

Item III.D.2.1, Radiological Monitoring of Effluents requires an NRR evaluation of modifying effluent monitoring design criteria based on TMI-2 and their experiences.

Item II.F.1(1), Noble Gas Effluent Monitor of Clarification of the TMI Action Plan Requirements (NUREG-0737) is being implemented to require adequate monitoring capability during accident conditions.

Subtask 2: Control of Radioactive Materials Released in Effluents

The purpose of this subtask was to review plant operating histories and prepare NUREG reports documenting the evaluations and recommending solutions to identified problems.

Various staff actions since 1978 (including NUREG reports and IE Bulletins) have resulted in the staff conclusion that no continuing need for additional staff guidance exists.

Subtask 3: Effects of Accidental Liquid Releases on Nearby Water Supplies

The purpose of this task was to perform a generic analysis of the consequences of liquid tank failures for those plants which received their license prior to issuance of the Standard Review Plan (SRP).

Experience in performing SRP analyses for newer plants has indicated that it is highly unlikely that radioactive concentrations in the nearest potable water supply could exceed 10 CFR Part 20 values.

Subtask 4: Performance of Solid Waste Systems

The purpose of subtask 4 was to perform an industry-wide survey to determine the extent to which power plants could process wastes and to develop plans for upgrading existing systems or adding new systems.

The NRC position relative to a requirement for an operable installed solid radwaste system has changed and, therefore, this subtask is no longer appropriate.

For the above reasons, Issue B-67 is being deleted from the NRR list of generic issues. Since Issue B-67 is being deleted, only Generic Issue A-02, "Appendix I" is appropriate to this topic.

The resolution of Issue A-02 is described in the Basis for Deletion for Topic XI-1. Topic XI-2 is being deleted from the SEP program for the same reasons.

TOPIC: XIII-1 Conduct of Operations

(1) Definition:

The organization, administrative controls, and operating experience will be reviewed. The existing organization and administrative controls will be compared with Standard Technical Specifications and guidance provided in Regulatory Guides 1.8 and 1.33 to determine the adequacy of the staff to protect the plant and to operate safely in routine, emergency, and long-term postaccident circumstances. The plant operating history will be reviewed to assess the combination of staff, operating controls and alarms, and administrative controls, in particular plant procedures, emergency planning, and offsite preparedness, to determine whether additional staff, qualifications, or administrative controls will be required for continued safe operation.

(2) Safety Objective:

To obtain reasonable assurance that the plant has enough people, with sufficient training and experience, and has administrative controls adequate to specify proper operation in routine, emergency, and postaccident conditions.

(3) Status:

Most of the older plants have staff members that meet the experience and educational requirements given in ANSI N18.1-1971 (endorsed by Regulatory Guide 1.8); however, a comparison against current criteria for the composite staff has not been made. These plants have provided training for subsequent plant staffs, and plant experience has, in general, demonstrated safe design and operation. Operating experience review is ongoing, and has been, in general, favorable. However, an analysis of this experience for trends, common elements, and potential hidden problems has not been systematically performed.

A review of Section VI of operating reactor licensees' Technical Specifications was begun in 1974 using Section VI of the Standard Technical Specifications (STS) as a model. As of September 1975, these reviews had been completed and the plants licensed prior to this time had been found to: (1) be acceptable and upgrading was not required, (2) require upgrading of only the reporting requirements, or (3) require improvement to be comparable to the STS model. Plants licensed after September 1975 have been reviewed against the STS model. Further review of Section VI, therefore, will not be required.

Emergency plans submitted at the operating-license stage complied with 10 CFR 50, Appendix E, 1970; however, these plans are not consistent with the guidance given in new Regulatory Guide 1.101, Revision 1, 1977.

(4) References:

1. Regulatory Guides
   1.8, "Personnel Selection and Training"
   1.33, "Quality Assurance Program Requirements (Operations)"
2. American National Standards Institute, ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel"
3. American National Standards Institute, ANSI N18.7-1972 Revised, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants"
4. Standard Technical Specifications, Section VI
5. 10 CFR Part 50, Appendix E
6. Regulatory Guide 1.101, Rev. 1, "Emergency Planning for Nuclear Power Plants"
7. Standard Review Plan, Section 13.3
8. NUREG 75/111, "Guide and Checklist for Development and Evaluation of State and Local Government Radiological Emergency Response Plans In Support of Fixed Nuclear Facilities," October 1975
9. Environmental Protection Agency, "EPA Manual of Protective Action Guides and Protective Action for Nuclear Incidents," September 1975
10. Memorandum of Understanding, NRR and Office of State Programs on State and Local Preparedness, March 10, 1977

(5) Basis for Deletion (Related TMI Task, USI, or Other SEP Topic):

(a) TMI Action Plan Task I.C.6, "Procedures for Verification of Correct Performance of Operating Activities," (NUREG-0737)

Under TMI Task I.C.6, a review of licensee procedures will be conducted to assure that an effective system of verifying the correct performance of operating activities exists. The purpose of this review is to provide a means of reducing human errors and improving the quality of normal operation. References cited for this review are ANSI Standard N18.7-1972 (ANS 3.2), "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," and Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operations)." These are the same references cited for Topic XIII-1.

(b) TMI Action Plan Task III.A.1, "Improve Licensee Emergency Prepared ness - Short-Term," and Task III.A.2, "Improving Licensee Emergency Preparedness - Long-Term" (NUREG-0660 and NUREG-0737)

Under Task III.A.1, a review of 10 CFR Part 50, Appendix E backfit requirements is being conducted in accordance with NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." The scope of NUREG-0654 covers Standard Review Plan, Section 13.3, and NUREG 75/111.

Regulatory Guide 1.101 has been deleted and has been superseded by an amended Appendix E to 10 CFR Part 50 (45 FR 55410, August 19, 1980). Under Task III.A.2, a review of licensee's emergency preparedness plans with respect to amended Appendix E will be conducted in accordance with NUREG-0654.

The evaluations required by TMI Tasks I.C.6, III.A.1, and III.A.2 are identical to SEP Topic XIII-1; therefore, this SEP topic has been deleted.

TOPIC: XIII-2 Safeguards/Industrial Security

(1) Definition:

Industrial security will be included under the scope of the operations review. Design features to assess the plant's capability to prevent sabotage and protect the operating unit(s) at dual or three-unit sites with unit(s) under construction will be included. Protective measures will be balanced against the sabotage threat. Fuel accountability will also be reviewed to assure that adequate inventory control procedures exist and the required records are kept.

(2) Safety Objective:

To determine that the plant has adequate security forces, design features, procedures and plans, and other administrative controls to meet the postulated sabotage threat. To assure that the fuel is adequately accounted for, that proper records are maintained, and the required reports are made.

(3) Status:

Each licensee currently has a security program and a fuel accountability program. Revised 10 CFR 73.55 has been published and submittals in accordance with its provisions were due May 25, 1977. These submittals are currently being evaluated.

(4) References:

1. 10 CFR Part 70
2. 10 CFR Part 73
3. Standard Technical Specifications, Section VI

TOPIC: XV-1 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of A Steam Generator Relief or Safety Valve

(1) Definition:

Review the assumptions, calculational models used and consequences of postulated accidents which involve an unplanned increase in heat removal. An excessive heat removal, that is, a heat removal rate in excess of the heat generation rate in the core, causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. If clad failure is calculated to occur, determine that offsite dose consequences are acceptable.

(2) Safety Objective:

To assure that pressures in the reactor coolant and main steam systems are limited in order to protect the reactor coolant pressure boundary from overpressurization and that fuel rod cladding failure as a result of departure from nucleate boiling ratio is limited.

(3) Status:

During each reload review by the staff, the previously determined limiting transient is reviewed to determine if new core parameters are more restrictive than the reference analysis parameter values.

(4) References:

Standard Review Plan, Sections 15.1.1 through 15.1.4

TOPIC: XV-2 Spectrum of Steam System Piping Failures Inside and Outside of Containment (PWR)

(1) Definition:

Review the assumptions, including use of nonsafety-grade equipment and concurrent steam generator or tube failure or blowdown of more than one steam generator, calculational models used, and consequences of postulated accidents which cause an increase in steam flow. The excessive steam flow reduces system temperature and pressure which increases core reactivity and can lead to a decrease of shutdown margin and departure from nucleate boiling ratio.

(2) Safety Objective:

To assure that (1) pressure in the reactor coolant and main steam lines is limited in order to protect the reactor coolant pressure boundary from overpressurization, (2) fuel damage is sufficiently limited so that the core will remain in place and intact with no loss of core cooling capability, (3) doses at the nearest exclusion area boundary are a small fraction of 10 CFR Part 100 guidelines, (4) ambient conditions do not exceed equipment qualification conditions (particularly nonsafety-grade equipment used to mitigate the accident), (5) the thermal and stress transients do not damage the reactor vessel, and (6) systems necessary for safe shutdown are not damaged by the accident.

(3) Status:

Investigation of the effects of high-energy line failures outside containment on other equipment was initiated as a generic issue in 1971 and all but a few facilities have been completed. New acceptance criteria have evolved during the review period. There was no similar investigation for failures inside containment. No reviews on operating plants of the effects on the reactor of concurrent steam generator or tube failure, or of blowdown of more than one steam generator have been performed.

(4) Reference:

Standard Review Plan, Section 15.1.5

TOPIC: XV-3 Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulatory Failure (Closed)

(1) Definition:

Review the assumptions, calculational models used, and consequences of postulated accidents which involve a decrease in secondary heat removal. The decrease in heat removal causes a suddent increase in system pressure and temperature.

(2) Safety Objective:

To assure that pressure in the reactor coolant and main steam systems is limited in order to protect the reactor coolant pressure boundary from overpressurization and that thermal margin for fuel integrity is maintained.

(3) Status:

The consequences associated with these transients are compared during each reload review to the consequences found to be acceptable during previous reload reviews.

(4) References:

Standard Review Plan, Sections 15.2.1 through 15.2.5

TOPIC: XV-4 Loss of Nonemergency AC Power to the Station Auxiliaries

(1) Definition:

Review the assumptions, calculational models used, and consequences of postulated accidents which involve the loss of nonemergency ac power (loss of offsite power or onsite ac distribution system) to station auxiliaries (for example, reactor coolant circulation pumps). This power loss will, within a few seconds, cause the turbine to trip and reactor coolant system to be isolated, which in turn causes the coolant pressure and temperature to increase.

(2) Safety Objective:

To assure that the pressure in the reactor coolant and main steam systems is limited in order to protect the reactor coolant pressure boundary from overpressurization and that thermal margin for fuel integrity is maintained.

(3) Status:

During each reload review by the staff, the previously determined limiting transient is reviewed to determine if new core parameters are more restrictive than the reference analysis parameter values.

(4) Reference:

Standard Review Plan, Section 15.2.6

TOPIC: XV-5 Loss of Normal Feedwater Flow

(1) Definition:

Review the assumptions, calculational models used, and consequences of the postulated loss of feedwater flow accidents, which cause an increase in coolant pressure and temperature.

(2) Safety Objective:

To assure that pressure in the reactor coolant and main steam systems is limited in order to protect the reactor coolant pressure boundary from overpressurization and that thermal margin for fuel integrity is maintained.

(3) Status:

The consequences associated with these transients are compared during each reload review to the consequences found to be acceptable during previous reload reviews.

(4) Reference:

Standard Review Plan, Section 15.2.7

TOPIC: XV-6 Feedwater System Pipe Breaks Inside and Outside Containment (PWR)

(1) Definition:

Review the assumptions, calculational models used, and consequences of postulated accidents which involve feedwater line breaks of different sizes. A feedwater line break, depending on size, may cause reactor system heatup (by reducing feedwater flow to the steam generator), or cooldown (by excessive energy discharge through the break).

(2) Safety Objective:

To assure that pressure in the reactor coolant and main steam systems is limited in order to protect the reactor coolant pressure boundary from overpressurization and that thermal margin for fuel integrity is maintained and that any radioactivity release would result in doses at the site boundary well within 10 CFR Part 100 guidelines.

(3) Status:

The identification of the most limiting transients and the consequences associated with these transients is evaluated during each reload review by the staff.

(4) Reference:

Standard Review Plan, Section 15.2.8

TOPIC: XV-7 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break

(1) Definition:

Review the assumptions, calculational models, and consequences of seizure of the rotor or break of the shaft of a reactor coolant pump in a PWR or recirculation pump in a BWR. These accidents result in a sudden decrease in core coolant flow and corresponding degradation of core heat transfer and, in a PWR, an increase in primary system pressure. If clad failure is calculated, determine that offsite consequences are acceptable.

(2) Safety Objective:

To assure that the consequences of a reactor coolant pump rotor seizure or reactor coolant pump shaft break are acceptable; that is, that no more than a small fraction of the fuel rods fail, that the radiological consequences are a small fraction of 10 CFR Part 100 guidelines, and that the system pressure is limited in order to protect the reactor coolant pressure boundary from overpressurization.

(3) Status:

Reviewed during each reload only if there is reason to believe that results would be different from the reference analysis; that is, only if a change in core parameters invalidates previous analyses.

(4) Reference:

Standard Review Plan, Section 15.3.3

TOPIC: XV-8 Control Rod Misoperation (System Malfunction or Operator Error)*

(1) Definition:

Review the licensee's description of rod position, flux, pressure, and temperature indication systems and the actions initiated by those systems which can mitigate the effects or prevent the occurrence of various misoperations. Review the descriptions of the input calculations and the calculational models used and the justification of their validity and adequacy. A transient of this type can result in achieving fuel melt temperatures and potential fuel damage.

(2) Safety Objective:

To assure that the consequences of this event do not exceed specified fuel design limits and that the protection system action be initiated automatically.

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*Reviewed for PWRs only; Standard Review Plan, Sections 15.4.1 and 15.4.2 cover BWRs and no additional areas considered.

(3) Status:

Reviewed during reload, Technical Specifications revised to compensate for changes in analytical results.

(4) Reference:

Standard Review Plan, Section 15.4.3

TOPIC: XV-9 Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate

(1) Definition:

Review BWRs for (1) startup of an idle recirculation pump and (2) a flow controller malfunction causing increased recirculation flow. Review PWRs with loop isolation valves for startup of a pump in an initially isolated inactive reactor coolant loop where the rate of flow increase is limited by the rate at which isolation valves open. For PWRs without loop isolation valves, review startup of a pump in any inactive loop. If clad failures are calculated, determine that offsite consequences are acceptable.

(2) Safety Objective:

To verify that the plant responds in such a way that the criteria regarding fuel damage and system pressure are met (that is, no more than a small fraction of the fuel rods fail, that radiological consequences are a small fraction of 10 CFR Part 100 guidelines, and that the system pressure is limited in order to protect the reactor coolant pressure boundary from overpressurization.)

(3) Status:

PWRs reviewed against the final safety analysis report, BWR reviewed at each reload, Technical Specifications required to preclude exceeding safety limits during transients.

(4) Reference:

Standard Review Plan, Sections 15.4.4 and 15.4.5

TOPIC: XV-10 Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)

(1) Definition:

Review the assumptions, calculational models used, and consequences of moderator dilution. An accident of this type could result in a departure from nucleate boiling and a loss of shutdown margin.

(2) Safety Objective:

To confirm that the plant responds to the events in such a way that the criteria regarding fuel damage and system pressure are met and adequate time allowed for the operator to terminate the dilution before the shutdown margin is reduced. (Reactor coolant pressure and main steam pressure should be limited in order to protect the reactor coolant pressure boundary from overpressurization.) (Operator action must be initiated within 30 minutes following this event if refueling, and within 15 minutes during other modes of operation.)

(3) Status:

Only reviewed during initial operating-license review and not thereafter. The consequences may not have been calculated in accordance with current practice.

(4) Reference:

Standard Review Plan, Section 15.4.6

TOPIC: XV-11 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (BWR)

(1) Definition:

Review the spectrum of misloading events analyzed to verify that the worst situation undetectable by incore instrumentation has been identified. This review will include an assessment of the plant's offgas and steam line radiation monitors to detect fuel damage and their capability to automatically isolate the offgas system when necessary.

(2) Safety Objective:

To assure that a misloaded assembly is detected and if undetected will not result in exceeding fuel safety limits or radioactive releases.

(3) Status:

Reviewed during reloads, Technical Specifications developed to limit consequences of worst misloaded assembly to small fraction of 10 CFR Part 100 guidelines. Technical Specifications setpoints for radiation monitors alarm/isolation signals have been found deficient and have been updated on a case-by-case basis for several plants.

(4) Reference:

Standard Review Plan, Section 15.4.7

TOPIC: XV-12 Spectrum of Rod Ejection Accidents (PWR)

(1) Definition:

Review the assumptions, calculational models used, and consequences, including radiological consequences, of PWR control rod ejection accidents, and review the Technical Specifications regarding control of reactivity worth and technical specifications on primary to secondary leakage. Ejection of a control element assembly from the core can occur if the control element drive mechanism housing or the nozzle on the reactor vessel head breaks off circumferentially. The ejection of a control element assembly by the reactor coolant system pressure can cause a severe reactivity excursion. This accident may result in high doses for those plants where fuel failures are postulated to occur as a result of the accident. This accident usually determines the maximum allowable steam generator leak rate.

(2) Safety Objective:

To ensure that if a control element assembly ejection occurs, core damage is minimal, no additional reactor coolant pressure boundary failures occur, the calculated radial average energy density is limited to 280 cals/gm at any axial fuel location in any fuel rod, and that the radiological consequences will not exceed appropriate limits.

(3) Status:

Releases through the containment and/or steam generator leaks are analyzed for current plants, but were not reviewed routinely for older plants. Many of the operating plants have no leak Technical Specifications or they are excessively high. During each reload by the staff, the previously determined limiting transient is reviewed to determine if the new ejected rod worth is more restrictive than the reference analysis values.

(4) References:

1. Standard Review Plan, Section 15.4.8
2. Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors"

TOPIC: XV-13 Spectrum of Rod Drop Accidents (BWR)

(1) Definition:

Review the assumptions, calculational models used, and consequences of BWR control rod drop accidents and review the Technical Specifications regarding control of rod activity worth. An uncoupled rod may hang up in the core when the control rod drive is withdrawn and drop later when the consequences of a rapid control rod withdrawal are most severe. An analysis of the radiological consequences from this accident will be included.

(2) Safety Objective:

To limit the effects of a postulated control rod drop to the extent that reactor coolant pressure boundary stresses are not exceeded and core damage is minimal. To assure that the radial average fuel rod enthalpy at any axial location in any fuel rod is limited to less than 280 cals/gm following the worst reactivity excursion and to assure that the radiological consequences do not exceed appropriate guidelines.

(3) Status:

The potential for and reactivity consequences of an accidental control rod drop are now routinely evaluated prior to issuance of an operating license and any time thereafter when changes could affect the accident results or probability of occurrence. Radiological consequences may not have been calculated in accordance with present practice.

(4) Reference:

Standard Review Plan, Section 15.4.9

TOPIC: XV-14 Inadvertent Operation of Emergency Core Cooling System and Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory

(1) Definition:

Review the assumptions, calculational models used, and consequences of actuation of the high pressure coolant injection system or faulty operation of the volume control system. The chemical and volume control system regulates both the chemistry and the quantity of coolant in the reactor coolant system. Changing the boron concentration in the reactor coolant system is a part of normal plant operation, compensating for long-term reactivity effects. Actuation of these systems could increase the volume of coolant within the reactor coolant pressure boundary (RCPB) causing a high water level, possible high power level, and high or low pressure. If clad failure is calculated, determine that offsite consequences are acceptable.

(2) Safety Objective:

To assure that water added to the RCPB does not cause transients that exceed RCPB pressure limits or result in unacceptable fuel damage. No activity is released during the transient, but the transient may subsequently result in increased radioactivity in gaseous releases during normal operation.

(3) Status:

This transient is now routinely analyzed prior to issuance of an operating license and any time thereafter when proposed changes would affect the transient results. Radiological consequences may not have been calculated in accordance with current practice.

(4) Reference:

Standard Review Plan, Section 15.5.1

TOPIC: XV-15 Inadvertent Opening of a PWR Pressurizer Safety/Relief Valve or a BWR Safety/Relief Valve

(1) Definition:

Review the assumptions, calculational models used, and consequences of inadvertent opening of a PWR pressurizer safety/relief valve or a BWR safety/relief valve. Loss of reactor coolant inventory and depressurizing action of the reactor coolant system can occur if the PWR pressurizer safety/relief valve or the BWR safety/relief valves open spuriously, or open when required but fail to reclose properly.

(2) Safety Objective:

To preserve fuel cladding integrity during reactor coolant system depressurization transients resulting from faulty operation of a relief or safety valve while at rated power.

(3) Status:

The transient is now evaluated prior to issuance of an operating license and any time thereafter when proposed changes could affect the transient results.

(4) References:

1. Standard Review Plan, Section 15.5.1
2. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants"

TOPIC: XV-16 Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment

(1) Definition:

Review the assumption, calculational models used, and radiological consequences of failure of small lines carrying primary coolant outside containment and review the Technical Specifications associated with primary coolant radioactivity concentrations, isolation valve closure times, and isolation valve leakage limits. In the event of a rupture of any component in the instrument lines outside primary containment, primary coolant and any radioactivity contained in the coolant or released to the coolant during the transient will be released if the instrument lines are connected to the reactor coolant pressure boundary. Primary coolant sample lines if broken outside primary containment can also allow coolant and radioactivity in the coolant to escape in the same manner. When these lines discharge to secondary containment, the integrity of the secondary containment and the efficiency of the filtration systems must be determined.

(2) Safety Objective:

To assure that any release of radioactivity to the environment is substantially below the guidelines of 10 CFR 100.

(3) Status:

The radiological consequences of small line breaks outside of primary containment have been evaluated routinely since 1970 prior to issuance of operating licenses, but have not always included the effects of iodine spikes during the depressurization transient.

(4) References:

1. Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment"
2. 10 CFR Part 50, Appendix A, GDC 55 and 56
3. Standard Review Plan, Section 15.6.2

TOPIC: XV-17 Radiological Consequences of Steam Generator Tube Failure (PWR)

(1) Definition:

Review the assumptions, calculational models used, and consequences of a steam generator tube failure with and without loss of offsite power and review the Technical Specifications associated with coolant activity concentrations. Steam generator tube failures allow escape of reactor coolant into the main steam system and to the environment. An analysis of the radiological consequences of this accident will be included.

(2) Safety Objective:

To assure that the plant responds in a proper manner to this accident, including appropriate operator actions, and to assure that radioactivity released following steam generator tube failure(s) is a small fraction of the 10 CFR 100 guidelines and within 10 CFR 100 for the case of a coincident iodine spike.

(3) Status:

The iodine release mechanism may not have been analyzed in accordance with present assumptions and methods for some of the older PWRs. Some operating plants do not have iodine activity limits in their Technical Specifications or have inappropriately high limits.

(4) References:

1. Standard Review Plan, Section 15.6.3
2. Regulatory Guide 1.5, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors"

TOPIC: XV-18 Radiological Consequences of Main Steam Line Failure Outside Containment

(1) Definition:

Review the assumptions, calculational models used, and consequences of failure of a main steam line outside containment and review the Technical Specifications associated with primary coolant activity concentrations and main steam isolation valve closure times.

(2) Safety Objective:

A steam line break outside containment allows radioactivity to escape to the environment. To limit the release of radioactivity to the environment to well within the guidelines of 10 CFR 100 in the event of a large steam line break, the primary coolant radioactivity must be appropriately limited by Technical Specifications.

(3) Status:

Some operating plants do not have appropriate coolant activity Technical Specifications.

(4) Reference:

Standard Review Plan, Section 15.6.4

TOPIC: XV-19 Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

(1) Definition:

Review the licensee's analyses of the spectrum of loss-of-coolant accidents (LOCAs) including break locations, break sizes, and initial conditions assumed, the evaluation model used, failure modes, radiological consequences, acceptability of auxiliary systems, functional capability of the containment, and the effects of blowdown loads. LOCAs are postulated breaks in the reactor coolant pressure boundary resulting in a loss of reactor coolant at a rate in excess of the capability of the reactor coolant makeup system. LOCAs result in excessive fuel damage or melt unless coolant is replenished.

(2) Safety Objective:

To assure that the consequences of loss-of-coolant accidents are acceptable; that is, that the requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50 are met, that the radiological consequences of a design basis loss-of-coolant accident from containment leakage and the radiological consequences of leakage from engineered safety features outside containment are acceptable, and the structural effects of blowdown are acceptable.

(3) Status:

Emergency core cooling system (ECCS) evaluation is a generic item which is currently under review or is complete for all operating reactors (La Crosse and San Onofre have stainless steel cores and have analyses completed to show conformance with the Interim Acceptance Criteria). Related generic items currently under review are reevaluations for increased vessel head fluid temperatures in W PWRs, effects of core flow on BWR LOCA analyses, GE ECCS input errors, and non-jet pump BWR core spray cooling coefficients. Radiological consequences are not routinely rereviewed.

(4) Reference:

Standard Review Plan, Section 15.6.5 and its Appendices

TOPIC: XV-20 Radiological Consequences of Fuel-Damaging Accidents (Inside and Outside Containment)

(1) Definition:

Review the assumptions, calculational models used, and consequences of postulated fuel damaging accidents inside and outside containment and review Technical Specifications associated with fuel handling and ventilation system and filter systems, including interlocks on fuel movement and damage from fuel cask drop and tipping. Include in the review the assumed activity available for release, decontamination factors, filter efficiencies, activity transport mechanisms and rates, ventilation system potential release pathways, and calculated doses.

(2) Safety Objective:

To assure that offsite doses resulting from fuel damaging accidents, resulting from fuel handling, or dropping a heavy load on fuel are well within the guideline values of 10 CFR Part 100.

(3) Status:

The radiological consequences of fuel handling accidents inside containment are currently being performed as a generic review for PWRs. The radiological consequences of fuel damaging accidents outside containment of operating plants are only evaluated if Technical Specifications are reviewed.

(4) References:

1. Standard Review Plan, Section 15.7.4
2. Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors"

TOPIC: XV-21 Spent Fuel Cask Drop Accidents

(1) Definition:

Review the potential for spent fuel cask drops, the damage which could result from cask drops, and the radiological consequences of a cask drop from fuel damaged within the cask under conditions exceeding the design basis impact on the cask.

(2) Safety Objective:

To assure that the damage to fuel within the casks and radiological consequences resulting from a cask drop are acceptable or that acceptable measures have been taken to preclude cask drops.

(3) Status:

Fuel cask drop analysis is a generic item which has been completed on some plants or is currently under review for all other operating reactors.

(4) References:

1. Standard Review Plan, Section 15.7.4
2. Regulatory Guide 1.25 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors"
3. NUREG-0328, "Regulatory Licensing: Status Summary Report" (Pink Book)

(5) Basis for Deletion (Related TMI Task, USI, or Other SEP Topic):

USI A-36, "Control of Heavy Loads Near Spent Fuel" (NUREG-0649)

The review criteria required by USI A-36 (Standard Review Plan, Section 15.7.5) are identical to the review criteria specified in the References of SEP Topic IX-2; therefore, this SEP topic has been deleted.

TOPIC: XV-22 Anticipated Transients Without Scram

(1) Definition:

Review the postulated sequences of events, analytical models, values of parameters used in the analytical models, and the predicted results and consequences of events in which an anticipated transient occurs and is not followed by an automatic reactor shutdown (scram). Analyses of the radiological consequences for these transients will be included. Failure of the reactor to shut down quickly during anticipated transients can lead to unacceptable reactor coolant system pressures and to fuel damage.

(2) Safety Objective:

To assure that the reliability of the reactor shutdown systems is high enough so that anticipated transient without scram (ATWS) events need not be considered or to assure that the consequences of ATWS events are acceptable; that is, that the reactor coolant system pressure, fuel pressure, fuel thermal and hydraulic performance, maximum containment pressure, and radiological consequences are within acceptable limits.

(3) Status:

ATWS is a generic topic currently under review to determine a position for all power reactors. BWR licensees have been requested to install reactor coolant pump trips as a short-term program measure. All licensees have submitted descriptions of the applicability of vendor generic ATWS reports for their plants. The schedule for review of Class C plants, which includes those plants designated for Phase II of SEP, has not yet been developed.

(4) References:

1. NUREG-0328, "Regulatory Licensing: Status Summary Report" (Pink Book)
2. WASH 1270, "Technical Report on Anticipated Transients Without Scram for Water-Cooled Power Reactors," September 1973
3. Standard Review Plan, Section 15.8 and Appendix

(5) Basis for Deletion (Related TMI Task, USI, or Other SEP Topic):

USI A-9, "Anticipated Transients Without Scram" (NUREG-0606)

The reference cited in this topic, that is, NUREG-0328, was the precursor of USI A-9. The evaluation required for USI A-9 is identical to SEP Topic XV-22; therefore, this SEP topic has been deleted.

TOPIC: XV-23 Multiple Tube Failures in Steam Generators

(1) Definition:

Assess the effects of multiple steam generator tube failures (ranging from leaks to double-ended ruptures) as a result of pressure differentials that may occur following a loss-of-coolant accident (LOCA), steam line break, or anticipated transient without scram (ATWS) events.

(2) Safety Objective:

Assure that the reflood of the core following a LOCA is possible and that the radiological consequences following these accidents are within the 10 CFR Part 100 guidelines.

(3) Status:

The consequences of multiple tube failures have not been analyzed for any plant at the licensing stage. Work has been done for some operating plants, but ultimate goals have yet to be set.

(4) References:

1. Prairie Island Nuclear Station, Docket Nos. 50-282 and 50-306
2. Turkey Point Plant, Docket Nos. 50-250 and 50-251
3. Surry Power Stations, Units 1 and 2, Docket Nos. 50-280 and 50-281

(5) Basis for Deletion (Related TMI Task, USI, or Other SEP Topic):

(a) USI A-3, A-4, A-5, "Westinghouse, Combustion Engineering, Babcock and Wilcox Steam Generator Tube Integrity" (NUREG-0649)

Two of the tasks of USI A-3, A-4, A-5 are as follows:

1. Analyses of LOCA with Concurrent Steam Generator Tube Failures
2. Analyses of Main Steam Line Break

The analyses required by these two tasks in USI A-3, A-4, A-5 cover two of the three events specified in the Definition.

(b) USI A-9, "Anticipated Transients Without Scram" (NUREG-0606)

Pressure differentials resulting from ATWS events have been determined to be no greater than those resulting from main steam line break events (NUREG-0460, Volume 2, Appendix V). The analysis for ATWS event is, therefore, covered under USI A-3, A-4, and A-5.

The evaluation required for USI A-3, A-4, A-5 is identical to SEP Topic XV-23; therefore, this SEP topic has been deleted.

TOPIC: XV-24 Loss of All AC Power

(1) Definition:

Review plant systems to determine that following loss of all ac power (onsite and offsite) the reactor is shut down and core cooling can be initiated. Loss of all ac power causes loss of most emergency equipment and instrumentation.

(2) Safety Objective:

To assure that with only dc power, equipment design, diversity, and operator action are sufficient to initiate core cooling within a short time period (typically 20 minutes).

(3) Status:

Not an explicit SRP topic. Availability of some ac power is assumed in all accident/transient analyses. Topic may be considered as an auxiliary fuel pump or reactor core isolation cooling pump diversity spinoff.

(4) Basis for Deletion (Related TMI Task, USI, or Other SEP Topic):

- USI A-44, "Station Blackout" (NUREG-0606)

  The problem description of USI A-44 is identical to the Definition of SEP Topic XV-24, and the review of USI A-44 would be the same as Topic XV-24; therefore, this SEP topic has been deleted.

TOPIC: XVI Technical Specifications

(1) Definition:

The existing Technical Specifications, associated with SEP topics, will be compared with the Standard Technical Specifications for deviations. Where significant differences exist, they will be identified and considered for upgrading. The bases for the specifications will be examined including trip setpoints and accounting for nuclear uncertainty. Where significant voids occur in existing specifications, appropriate values will be identified and considered for upgrading.

(2) Safety Objective:

To assure that the safety limits and operational safety measures are sufficiently specified for the plant to minimize the probability of accidents that could result from equipment failure, misoperation, or human error.

(3) Status:

See Topic XIII-1, "Conduct of Operations" for Section VI status. The other sections of the Technical Specifications are reviewed only to the extent that reloads, license amendments, or generic problems require.

(4) References:

1. Standard Technical Specifications; Regulatory Guide 1.8, "Personnel Selection and Training," and Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operations)"
2. Standard Review Plan
3. Regulatory Guide 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Chapter 16
4. 10 CFR Part 50, Section 50.36

TOPIC: XVII Operational Quality Assurance Program

(1) Definition:

Review the Quality Assurance (QA) Program with respect to safe and reliable operation of the plant.

(2) Safety Objective:

Since 1973, significant new guidance for operational QA programs in the form of Regulatory Guides and WASH documents has been issued describing how to meet the criteria of 10 CFR Part 50, Appendix B. The objective of this guidance is to assure that operation, maintenance, modification, and test activities do not degrade the capability of safety-related items to perform their intended functions.

(3) Status:

Generic review for compliance with current standards is under way. As of May 1977, 50 of the 63 operating plants have QA programs which meet current criteria. The 13 remaining plants are currently under review, with an estimated completion date of July 1977.

(4) References:

1. 10 CFR Part 50, Appendix B
2. WASH-1283, Revision 1, "Guidance on Quality Assurance Requirements During Design and Procurement Phase of Nuclear Power Plants," May 24, 1974
3. WASH-1284, "Guidance on Quality Assurance Requirements During the Operations Phase of Nuclear Power Plants," October 26, 1973

4. WASH-1309, "Guidance on Quality Assurance Requirements During the Construction Phase of Nuclear Power Plants," May 10, 1974
5. American National Standards Institute, ANSI N18.7-1976, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," February 19, 1976

U.S. Nuclear Regulatory Commission reports cited under "Basis for Deletion" include:

NUREG-75/111	Guide and Checklist for Development and Evaluation of State and Local Government Radiological Emergency Response Plans in Support of Fixed Nuclear Facilities" (Reprint of WASH-1293), Oct. 1975.
NUREG-0153	"Staff Discussion of 12 Additional Technical Issues Raised by Responses to November 3, 1976 Memorandum from Director, NRR, to NRR staff," 1976.
NUREG-0313	"Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," July 1977.
NUREG-0328	"Regulatory Licensing: Status Summary Report" (Pink Book).
NUREG-0371	"Approved Category A Task Action Plans," Nov. 1977.
NUREG-0410	"NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants, Report to Congress," Dec. 1977.
NUREG-0460	"Anticipated Transients Without Scram for Light Water Reactors," Vol. 2, Apr. 1978.
NUREG-0471	"Generic Task Problem Descriptions - Category B, C, and D Tasks," Sept. 1978.
NUREG-0484	"Methodology for Combining Dynamic Responses," May 1980.
NUREG-0510	"Identification of Unresolved Safety Issues Relating to Nuclear Power Plants--A Report to Congress 1979," Jan. 1979.
NUREG-0554	"Single-Failure-Proof Cranes for Nuclear Power Plants," May 1979.
NUREG-0577	"Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports," Sept. 1979.
NUREG-0606	"Unresolved Safety Issues Summary," issued quarterly.
NUREG-0609	"Asymmetric Blowdown Loads on PWR Primary Systems, Resolution of Generic Task Action Plan A-2," Jan. 1981.
NUREG-0649	"Task Action Plan for Unresolved Safety Issues Related to Nuclear Power Plants," Feb. 1980.

NUREG-0654	"Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Feb. 1980.
NUREG-0660, Rev. 1	"NRC Action Plan Developed as a Result of the TMI-2 Accident," Vols. 1 and 2, May 1980
NUREG-0691	"Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors," Sept. 1980.
NUREG-0705	"Identification of New Unresolved Safety Issues Relating to Nuclear Power Plants," Mar. 1981.
NUREG-0737	"Clarification of TMI Action Plan Requirements," Nov. 1980.
NUREG-0800	"Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1981 (formerly NUREG-75/087).
NUREG/CR-1321	"Final Report - Phase I. Systems Interaction Methodology Applications Program," Apr. 1980.

APPENDIX B

SEP TOPICS DELETED BECAUSE THEY ARE COVERED BY A TMI TASK, UNRESOLVED SAFETY ISSUE (USI), OR OTHER SEP TOPIC[1,2]

[1]See "Basis for Deletion" in Appendix A under applicable SEP topic.
[2]Letter from G. C. Lainas (NRC) to all SEP licensees, Subject: Deletion of Systematic Evaluation Program Topics Covered by Three Mile Island NRC Action Plan, Unresolved Safety Issues, or Other SEP Topics, May 1981.

SEP Topic No.	SEP Title	TMI, USI, or SEP No.	TMI, USI, or SEP Title
II-2.B	Onsite Meteorological Measurements Program	TMI II.F.3 TMI III.A.1	Instrumentation for Monitoring Accident Conditions Improve Licensee Emergency Preparedness - Short Term
II-2.D	Availability of Meteorological Data in the Control Room	TMI II.F.3 TMI III.A.1 TMI I.D.1	Instrumentation for Monitoring Accident Conditions Improve Licensee Emergency Preparedness - Short Term Control Room Design Reviews
III-8.D	Core Supports and Fuel Integrity	USI A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant System
III-9	Support Integrity	USI A-12 USI A-7 USI A-24 USI A-46 SEP III-6 SEP V-1	Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports Mark I Containment Long-Term Program Environmental Qualification of Safety-Related Equipment Seismic Qualification of Equipment in Operating Plants Seismic Design Considerations Compliance With Codes and Standards (10 CFR Part 50, Section 50.55a)
III-11	Component Integrity	USI A-46 USI A-2 SEP III-6	Seismic Qualification of Equipment in Operating Plants Asymmetric Blowdown Loads on Reactor Primary Coolant Seismic Design Considerations
III-12	Environmental Qualification of Safety-Related Equipment	USI A-24	Qualification of Safety-Related Equipment
V-13	Waterhammer	USI A-1	Waterhammer
VI-2.A	Pressure-Suppression-Type BWR Containments	USI A-7	Mark I Containment Long-Term Program
VI-2.B	Subcompartment Analysis	USI A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant System
VI-5	Combustible Gas Control	TMI II.B.7 USI A-48	Analysis of Hydrogen Control Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment
VI-7.E	Emergency Core Cooling System Sump Design and Test for Recirculation Mode Effectiveness	USI A-43	Containment Emergency Sump Reliability
VI-8	Control Room Habitability	TMI III.D.3.4	Control Room Habitability Requirements
VII-4	Effects of Failure in Nonsafety-Related Systems on Selected Engineered Safety Features	USI A-47 USI A-17	Safety Implications of Control Systems Systems Interactions in Nuclear Power Plants
VII-5	Instruments for Monitoring Radiation and Process Variables During Accidents	TMI II.F.1 TMI II.F.2 TMI II.F.3	Additional Accident Monitoring Instrumentation Identification of and Recovery From Conditions Leading to Inadequate Core Cooling Instruments for Monitoring Accident Conditions
IX-2	Overhead Handling Systems (Cranes)	USI A-36	Control of Heavy Loads Near Spent Fuel Pool
XIII-1	Conduct of Operations	TMI I.C.6 TMI III.A.1 TMI III.A.2	Procedures for Verification of Correct Performance of Operating Activities Improve Licensee Emergency Preparedness - Short-Term Improving Licensee Emergency Preparedness - Long-Term
XV-21	Spent Fuel Cask Drop Accidents	USI A-36	Control of Heavy Loads Near Spent Fuel Pool
XV-22	Anticipated Transients Without Scram	USI A-9	Anticipated Transients Without Scram
XV-24	Loss of All AC Power	USI A-44	Station Blackout

APPENDIX C

PLANT-SPECIFIC SEP TOPICS DELETED, REFERENCE LETTER, AND REASON FOR DELETION

SEP Topic No.	SEP title	Date of letter	Reason for deletion of topic
III-3.B	Structural and Other Consequences (e.g., Flooding of Safety-Related Equipment in Basements) of Failure of Underdrain Systems	11/16/79	Not applicable to site because site does not have a system whose function is to lower the groundwater table.
III-7.A	Inservice Inspection, Including Prestressed Concrete Containments with Either Grouted or Ungrouted Tendons	11/16/79	Not applicable to this unit's containment design.
III-7.C	Delamination of Prestressed Concrete Containment Structures	11/16/79	Not applicable to this unit's containment design.
III-8.B	Control Rod Drive Mechanism Integrity	9/11/80	Review published as NUREG-0479, "Report on BWR Control Rod Drive Failures."
III-10.B	Pump Flywheel Integrity	11/16/79	Not applicable to BWRs.
V-1	Compliance With Codes and Standards	11/27/81	Reviewed under inservice inspection/ inservice test program.
V-2	Applicability of Code Cases	11/16/79	Not applicable at this time; to be reviewed for any future modifications using references to Code Cases.
V-3	Overpressurization Protection	11/16/79	Not applicable to BWRs based on operating experience.
V-7	Reactor Coolant Pump Overspeed	11/16/79	Not applicable to BWRs.
V-8	Steam Generator Integrity	11/16/79	Not applicable to BWRs.
V-9	Reactor Core Isolation Cooling System (BWR)	11/16/79	Not applicable to this facility design.
VI-2.C	Ice Condenser Containment	11/16/79	Not applicable to this unit's containment design.
VI-7.A.1	Emergency Core Cooling System Reevaluation To Account for Increased Reactor Vessel Upper-Head Temperature	11/16/79	Not applicable to BWRs.
VI-7.A.2	Upper Plenum Injection	11/16/79	Not applicable to BWRs.
VI-7.B	Engineered Safety Feature Switchover From Injection to Recirculation Mode (Automatic Emergency Core Cooling System Realignment)	11/16/79	Not applicable to BWRs.
VI-7.C.3	Effect of PWR Loop Isolation Valve Closure During a Loss-of-Coolant Accident on Emergency Core Cooling System Performance	11/16/79	Not applicable to BWRs.
VI-7.F	Accumulator Isolation Valves Power and Control System Design	11/16/79	Not applicable to BWRs.
VI-9	Main Steam Line Isolation Seal System (BWR)	11/16/79	Not applicable to this facility design.
VII-7	Acceptability of Swing Bus Design on BWR-4 Plants	11/16/79	Not applicable to this facility design.
IX-4	Boron Addition System (PWR)	11/16/79	Not applicable to BWRs.
X	Auxiliary Feedwater System	11/16/79	Not applicable to BWRs.
XI-1	Appendix I	12/4/81	Being resolved under generic activities A-02, "Appendix I," and B-35, "Confirmation of Appendix I Models." (See "Basis for Deletion" in Appendix A under Topic XI-1.)
XI-2	Radiological (Effluent and Process) Monitoring Systems	12/4/81	Being resolved under generic activities A-02, "Appendix I." (See "Basis for Deletion" in Appendix A under Topic XI-2.)

SEP Topic No.	SEP title	Date of letter	Reason for deletion of topic
XV-2	Spectrum of Steam System Piping Failures Inside and Outside Containment (PWR)	11/16/79	Not applicable to BWRs.
XV-6	Feedwater System Pipe Breaks Inside and Outside Containment (PWR)	11/16/79	Not applicable to BWRs.
XV-10	Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)	11/16/79	Not applicable to BWRs.
XV-12	Spectrum of Rod Ejection Accidents (PWR)	11/16/79	Not applicable to BWRs.
XV-17	Radiological Consequences of Steam Generator Tube Failure (PWR)	11/16/79	Not applicable to BWRs.
XV-23	Multiple Tube Failures in Steam Generators	11/16/79	Not applicable to BWRs.
XVI	Technical Specifications	11/05/80	Will be addressed after completion of the integrated assessment.

APPENDIX D

PROBABILISTIC RISK ASSESSMENT STUDY*,**

*"Risk Based Categorization of Dresden-2 SEP Issues" (Sandia National Laboratories), October 5, 1982.

**The modified fault trees are not reproduced in this report because of their bulk. A copy of the fault trees can be obtained by request.

Risk-Based Categorization of Dresden-2 SEP Issues

SAND82-2031

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EXECUTIVE SUMMARY

This is an Executive Summary of the report, "Risk-Based Categorization of Dresden-2 SEP Issues." Refer to the main report for the details of the analysis we have used to classify the Dresden-2 SEP issues with respect to their importance to risk. These classifications have been performed using probabilistic risk assessment (PRA) techniques.

The issues have been examined from the perspective of the impact their resolution would have on risk from the plant. The classifications are based on the criteria given in Table Ex-1. Following are discussions of each issue, their classifications based on these criteria, and the supportive results of our analysis which were judged by these criteria.

The Dresden-2 fault trees referred to in Table Ex-1 were constructed by modifying the IREP Millstone-1 system fault trees to represent the failures of the Dresden-2 systems. The methodology adopted in this study was to examine the impact of each issue on the systems it affects and assess the importance of the issue by qualitative consideration of the Dresden-2 fault trees, aided by the results and insights of other PRAs. For each issue, we estimated the impact its resolution would have on the modified Millstone-1 (Dresden-2) fault trees and thus the impact on the risk at Dresden-2. The "dominance" of a fault tree indicates whether that fault tree would appear in the dominant accident sequences.

Table Ex-2 gives the results of the classification of the issues as high, medium, or low importance to risk. The numbers denote the issues.

TABLE Ex-1

Criteria for Classification of Issues

Classification	Criterion
High	Resolution of issue dominates value of the top event of a dominant Dresden-2 fault tree or dominant sequence event.
Medium	Resolution of issue impacts but does not dominate value of top event of dominant fault tree or dominant sequence event.
Low	Resolution of issue has no impact on value of top event of dominant fault tree or dominant sequence event.

TABLE Ex-2

Issue Classification
(Importance to Risk)

High

VIII-3.A	Battery Testing
VIII-3.B	DC Bus Instrumentation

Medium

III-10.A	Thermal Overload Trips
V-11.B	Shutdown Cooling Interlock

Low

III-5.B	Pipe Break Outside Containment
III-8.A	Loose Parts
V-5	RCPB Leakage Detection
V-11.A	RWCU LOCA*
VI-4	Containment Penetrations
VI-6	Containment Leak Testing
VI-7.C.1	Electrical Distribution
VI-10.A	Response Time Testing
VI-10.B	DC Buses in Parallel, Fuel Oil
VII-1.A	RPS Isolation
VII-3	Shutdown Procedures
VIII-2	Diesel Trips
IX-5	Ventilation Systems
XV-1	PCS Transients
XV-16 }	Radiological Consequences
XV-18 }	Of Noncore-Melt Events

*Issue low importance if pressure relief valve is adequately sized. Issue high importance if not adequately sized.

III-5.B Pipe Break Outside Containment

We analyzed the risk importance of the various postulated pipe breaks as unisolatable LOCAs. We did not attempt to investigate the effects on other systems due to the physical effects of each pipe break. The actual number of lines of concern is small. We determined that the LOCA frequencies of these pipe breaks are all less than $\sim 2 \times 10^{-7}$/yr. Even if all these events led to core melt with release, the much higher frequencies of other core melt sequences coupled with the virtual certainty of containment failure in any case after core melt makes the contribution of these LOCAs to total risk negligible. The pipe break frequency is small enough that our conclusion also applies to the importance of the physical effects associated with the pipe break. That is, we assumed core melt following <u>every</u> break and showed that this was of negligible importance. Thus we classify this issue's importance to risk as low.

III-8.A Loose-Parts Monitoring and Core Barrel Vibration Monitoring

A loose-parts monitoring system as required by Regulatory Guide 1.133 does not exist at Dresden-2. Loose parts can cause transient events by causing damage within the reactor coolant system. However, the historical transient rate is high enough with a negligible contribution to that rate by loose parts to ensure that eliminating the loose-parts-induced transients will have no effect on risk.* This issue has, therefore, been classified as low importance to risk.

*Memo from L. S. Rubenstein to S. H. Hanauer, Director, Division of Safety Technology, and D. G. Eisenhut, Director, Division of Licensing, May 6, 1982, Subject: CRGR Briefing on Loose Parts Detection Program.

III-10.A Thermal Overload Protection for Motors of Motor-Operated Valves

Many systems contain motor-operated valves which do not have their thermal overload protection bypassed during accident conditions. If a spurious trip by the thermal overload occurs when a valve is required to operate it can degrade the safety function of the system containing the valve. Also, the NRC recommends replacing the torque switches with limit switches for valve protection. Our analysis showed that a single valve can have its unavailability reduced by about 14 percent by eliminating spurious thermal overload trips by bypassing the thermal overload protection. This effect is not a large one by PRA standards, but since many valves are affected this issue should impact slightly the value of the top event of several dominant event trees. Replacing the torque switches with limit switches would not improve the reliability of valve protection. Thus we classify the importance to risk of the issue as medium.

V-5 Reactor Coolant Pressure Boundary (RCPB) Leakage Detection

The Dresden-2 RCPB leakage detection systems cannot detect a 1-gallon per minute (gpm) leakage in 1 hour as required, but can detect 1 gpm in 8 hours. The hypothesis is that small LOCAs will begin as leaks (leak-before-break) and improved leakage detection can lead to prevention of some LOCAs. There are many unknowns which affect the analysis, perhaps the most important of which is the time it would take a 1 gpm leak to become a LOCA. Dresden-2 does not have to shut down, by Technical Specifications, until a 5 gpm leakage is discovered. It is unlikely that actions would be taken to prevent a LOCA even if a 1 gpm leak were discovered in 1 hour with improved detection. If the Technical Specifications were changed to require shutdown at 1 gpm, the small LOCA frequency could be affected under certain assumptions. The alternate failure mode where a pressure transient could cause a leak (which would otherwise not grow) to become a break if the transient occurred before the leak was detected, and the plant shutdown, was not analyzed. In addition, the high energy pipe break aspects (systems interactions) were not considered because PRAs do not assume that LOCAs cause any additional failures. In any case, PRAs of BWRs show few LOCA-initiated dominant accident sequences and, in particular, the results of the IREP Millstone-1 PRA and our assessment of the Dresden-2 fault trees give no LOCAs as dominant accidents. Thus we classify this issue as low importance to risk.

V-11.A Requirements for Isolation of High and Low-Pressure Systems

The Reactor Water Cleanup System (RWCU) should have independent interlocks on the suction valves to prevent an interfacing systems LOCA (outside containment). We examined the Dresden-2 RWCU system and calculated the present frequency of interfacing systems LOCAs through this system. The frequency through the suction line is about 2.5 x 10^{-7}/yr, assuming that the pressure relief valve is sufficiently sized. The Millstone-1 IREP PRA truncated consideration of core melt sequences at 10^{-6}/yr. This removed any sequence with a smaller frequency from further consideration and, in fact, the dominant sequences had much higher frequencies. Thus even if RWCU interfacing system LOCAs needed no other failures to produce a core melt, eliminating them would have a negligible effect on risk. The fact that the containment has already been bypassed in an interfacing system LOCA does not increase its importance in this case because, at Dresden-2, overpressure failure of containment is virtually ensured after any core melt (see Issue VI-4). (If the pressure relief valve is not sufficiently sized, this LOCA frequency becomes very important, ~2.4 x 10^{-3}/yr. This would result in this issue being classified as high importance to risk.) In addition, opening the relief valve can be considered a large LOCA event, here with frequency 2.4x10^{-3}/yr. This large LOCA frequency would not significantly contribute to the core melt frequency or risk (i.e., increasing the large LOCA frequency to this value would not increase any large LOCA induced accident sequence frequency enough to become a dominant sequence). Thus we classify this issue as low importance to risk.

V-11.B RHR Interlock Requirements

The shutdown cooling system is designed for full reactor pressure, but less than full reactor temperature. Therefore, system interlocks are based on temperature requirements. Current licensing criteria for the interlocks are not met since there are no testing requirements. If we assume that exceeding design temperature will always fail shutdown cooling, and that the temperature interlock is presently never tested, resolution of this issue would decrease the unavailability of shutdown cooling by about 15%. Since shutdown cooling is an important system in the dominant accident sequences identified at Millstone-1 and expected for Dresden-2, we classify this issue's importance to risk as medium.

VI-4 Containment Isolation Systems

VI-6 Containment Leak Testing

These issues address the adequacy of containment integrity during accident conditions. Because of the small size and low design pressure of the Dresden-2 containment, the pressure generated by steam and noncondensible gases during a core melt will fail the containment if no other failure mechanism occurs first. The overwhelmingly dominant portion of the risk from nuclear power plants is from core melt accidents, not other (low consequence) releases such as those due to non-core-melt accidents which result in relatively low (compared to core melt) doses to the public. Because of the characteristics and relative consequences of leakage releases and containment ruptures by overpressure, no benefit can be achieved by increasing the reliability of isolation of the containment since it will fail by overpressure anyway. Thus we classify these issues' importance to risk as low.

VI-7.C.1 Appendix K--Electrical, Instrumentation, and Control (EIC) Re-reviews

The portion of this issue that we analyzed deals with creating Technical Specifications to ensure that the reliability of ac and dc power is not degraded by operator actions. That is, the two 480-V switchgears 28 and 29 can be connected, tying the two ac power trains together. In addition, the instrumentation and control power for diesel generator 2/3 can be connected to the Unit 3 125-V distribution panel, which receives its power from the Unit 2 Division II power source. Procedures, to limit the conditions of concern, which conform to the proposed Technical Specifications already exist. Based on discussions with plant personnel and on the above mentioned procedures, these Technical Specifications would not change what is actually done at the plant, but enforce it. Thus we classify this issue's importance to risk as low.

VI-10.A Testing of Reactor Trip System and Engineered Safety Features, Including Response Time Testing

Periodic system response time measurements are not required by the technical specifications. In PRA analyses, timing of system action is relatively unimportant because the time periods measured by response time testing are very short. That is, from a risk perspective, it is generally acceptable if a system performs its function within a few tens of minutes from the time it is demanded, not within seconds as measured by response time testing. Failure of the system to operate at all is discovered by functional testing which is performed as required, and response time testing would not improve system unavailability. Thus we classify this issue's importance to risk as low.

VI-10.B Electrical, Instrumentation, and Control Portions of Shared Systems

There are three considerations for this issue: 1) the DC batteries can be operated in parallel, degrading their reliability, 2) there is no Technical Specification to prevent unit operation while the diesel generator 2/3 normal/bypass switch is in bypass, and 3) additional onsite fuel oil storage should be provided. Our analysis indicates that the probability of actually operating the batteries in parallel, leading to failure, is very small, $\sim 1.5 \times 10^{-9}$. In addition, the plant is always operated with the normal/bypass switch in normal and creating a Technical Specification will not change this. Finally, there is sufficient fuel oil for 2 days' operation of the diesels and, from a PRA perspective, this is more than adequate. Thus we classify this issue's importance to risk as low.

VII-1.A Isolation of Reactor Protection System (RPS) from Nonsafety Systems

There are no isolation devices between the nuclear flux monitoring systems and the process recorders for these systems where such devices are required and the isolation between the APRM System and the plant process computer may be inadequate. A fault in the nonsafety part of the nuclear flux monitoring channel or APRM could fail the high neutron flux signal or APRM. Even if this failure is assigned a probability of 1.0, the RPS failure probability from the Dresden-2 fault trees is totally dominated by common mode mechanical faults and eliminating the isolation problem has no effect on RPS unavailability. Thus we classify this issue as low importance to risk.

VII-3 Systems Required for Safe Shutdown

It is required that the plant can be taken from normal operating conditions to cold shutdown using only safety-grade systems, assuming single failure and utilizing either onsite or offsite power. Examination of plant procedures showed that provisions for shutting the plant down using both safety and nonsafety grade equipment were extensive enough to allow the operator to use alternate cooldown methods. It is extremely unlikely, from a PRA perspective, that only safety grade equipment would be available during an event and providing procedures utilizing only safety grade equipment would not improve shutdown system reliability. Thus we classify this issue's importance to risk as low.

VIII-2 Onsite Emergency Power Systems--Diesel Generators

One trip of the diesel generators does not meet the reliability standards for not being bypassed during accident conditions, but is not bypassed. This reduces the reliability of the diesel generators because this trip could spuriously make the diesel generators unavailable when needed. By examining Licensee Event Reports of diesel generator trips during tests (when all trips are not bypassed), we estimate that the one trip in question contributes, at most, about 2 percent to the unavailability of one diesel generator. Examination of the overall emergency ac power Dresden-2 fault tree showed that eliminating this small contribution from each diesel generator would have a negligible effect on the overall unavailability of emergency ac power.* Thus we classify this issue as low importance to risk.

*Emergency ac is an extremely important system from a risk perspective, and improving the system reliability would affect risk, but the impact of this issue is negligible.

VIII-3.A Station Battery Capacity Test Requirements

The station battery capacity tests do not conform to current licensing requirements. No periodic battery service tests are required and the load discharge test does not appear to verify the required 80 percent capacity. For calculational purposes, we assumed that all battery testing to date has been ineffective. This is an extremely conservative assumption. If the testing is implemented, we calculate that the battery failure rate can be reduced, at most, by a factor of 15. Since dc power does appear in dominant Dresden-2 fault trees, we classify this issue as high importance to risk.

VIII-3.B DC Power System Bus Voltage Monitoring and Annunciation

Dresden-2 has no control room indication of battery voltage, battery current, battery charger current, or breaker/fuse status. The concern is that, without adequate instrumentation, battery failures could go unnoticed between tests and the dc power system would be unavailable during an accident. From Licensee Event Reports, it is seen that approximately half the battery faults are detectable by instrumentation and half remain undetected until battery tests. An instrument detectable fault is repaired immediately, and so contributes little to unavailability of dc power. We calculated the unavailability of a dc bus with and without instrumentation which would detect those battery faults which are detectable. Installing such instrumentation at Dresden-2 would decrease the unavailability of a dc bus by about a factor of 5 assuming adequate battery testing. Since dc power would appear in some dominant accident sequences and resolution of this issue would have a significant impact on the value of the top event of the dc power fault tree, we classify this issue as high importance to risk.

IX-5 Ventilation Systems

There are three concerns from this issue: 1) adequacy of room cooling of emergency systems, 2) spread of radioactivity in the reactor building preventing operator entry, and 3) ventilation of the battery room. Studies performed for the IREP Millstone-1 PRA showed that room cooling was not an issue. No failures of emergency systems due to failures of room cooling were identified. For the second concern, by the time radioactivity from core damage is present in the reactor building, it is too late for the operator to perform recovery actions (requiring building access) to prevent core melt. It was outside the scope of this study to attempt to analyze the likelihood and consequences of loss of ventilation in the battery room, where hydrogen buildup during charging of the batteries could cause a detonation. Thus we classify this issue's importance to risk as low.

XV-1 Decrease in Feedwater Temperature, Increase in Feedwater Flow, and Increase in Steam Flow

Failure of the feedwater controller to maximum demand results in an increase in reactor power and vessel inventory. This event would be more severe with the turbine bypass system unavailable. Limitations to either reactor power or minimum critical power ratio would be required in the Technical Specifications for the case where the turbine bypass is found to be inoperable. The risk significance of this issue is that if the turbine bypass is inoperable, any transient will occur with the power conversion system (PCS) unavailable as a mitigating system. This cause of unavailability of the PCS was analyzed in the IREP Millstone-1 PRA, and it was found to be negligible. That is, the historical rate of turbine bypass failure has been small enough compared to other causes of loss of the PCS that even if the proposed limitations on reactor operation with the turbine bypass unavailable prevented transients under that condition, the effect on the overall transient rate with loss of the PCS would be negligible. Thus, we classify this issue's importance to risk as low.

XV-16 Radiological Consequences of Small Lines Carrying Primary Coolant Outside Containment

XV-18 Radiological Consequences of a Main Steam Line Failure Outside Containment

These two issues address exceeding 10 CFR Part 100 doses during events which do not lead to core melt. PRAs have shown that the overwhelming portion of the risk from nuclear plants is from core melt accidents, not other (small consequence) events. Thus we classify the importance of these issues to risk as low.

I. Introduction

This main report will present the analysis and results for the risk-based categorization of issues identified by the USNRC Systematic Evaluation Program (SEP) for the Dresden-2 nuclear power plant.

Section II will discuss the methodology, Section III will present our results for Dresden-2, Section IV will give the analysis performed for each Dresden-2 SEP issue, and Section V will give the IREP Millstone-1 fault trees as modified to represent Dresden-2.

A discussion of the analysis and results for each issue is given in the Executive Summary of this report.

II. Methodology for Categorization of Dresden-2 SEP Issues

The United States Nuclear Regulatory Commission (USNRC) Systematic Evaluation Program (SEP) is identifying deviations from current licensing requirements for older nuclear power plants. This project evaluates those issues which are amenable to study by probabilistic risk assessment (PRA) techniques. The result of this evaluation is the categorization of these issues by the impact their resolution would have on risk. This categorization will be used as input to the USNRC decisions on what hardware and procedure changes will be required for the nuclear plants as the product of the SEP.

Not all of the issues identified are easily addressed by well-defined PRA techniques. In particular, issues which address the ability of the power plant to safely deal with events for which the frequency and/or effects on plant systems are unknown are not evaluated in this study. PRA examines accident scenarios for which the initiating event frequencies are relatively well known and probabilities of system failures are estimated by detailed consideration of system configuration, random component failures, and system interactions. Thus the issues evaluated are those which address systems or plant features during normal operation or accident situations of relatively well-known frequency where those systems or plant features may be demanded.

Issues excluded are those dealing with seismic, tornado, or flooding events for which the frequency of an event of given severity, or any such event, is not well known. Also excluded are issues dealing with high energy line breaks, where it is not the frequency, but the effects on systems, which are not known. Treating these issues in the framework of PRA would generally be at the edge of the state-of-the-art (since event frequencies, etc., are not well known) and thus our confidence in the risk-based categorization of these issues would be less than for the results of our analysis of those issues which fit well into present PRA considerations.

Also not evaluated are issues dealing with the possible detection of weakening of the containment structure. In general, the actual strengths of containments are not known, although an estimation of the actual failure pressure is used in PRA calculations. If the actual changes in containment strength being considered were known, this issue could be addressed.

The PRA performed for the Millstone-1 nuclear power plant as part of the Interim Reliability Evaluation Program (IREP) was modified to represent Dresden-2. We modified the Millstone-1 system fault trees to represent the failures of the Dresden-2 systems. These modified fault trees and the changes in the Millstone data base necessary to solve them are included in this report in Section V.

The method adopted in this study was to examine the impact of each issue on the systems they affect and assess the importance of the issue by qualitative consideration of the Dresden-2 fault trees we developed, aided by the results and insights of other PRAs.

For each issue, we consider the impact its resolution would have on the calculation of risk with the modified Millstone fault trees and thus the risk at Dresden-2. That is, we assess the impact on the top event of each Dresden-2 fault tree (or event in any sequence) of each issue. This sometimes required developing further fault trees to incorporate the effect of each issue. For example, issues which impacted the failure rates of basic events or components required further development since the original fault tree ended at that level.

If we could ascertain no impact on the top event of any dominant fault tree (or event in any dominant sequence) due to resolution of an issue, we classified the issue's importance as low. If the resolution of the issue affects but does not dominate the fault tree (or event), the issue was classified as medium importance. If the resolution of the issue dominates the value of the top event of any dominant fault tree (or event), the issue's importance was classified as high.

This ranking includes our determination of whether a given Dresden-2 fault tree or event would appear in dominant accident sequences based on experience with solving the similar Millstone-1 fault trees and the results of PRAs of other BWRs.

The overall study methodology is given in flowchart form in Figure 1. The importance of an issue is determined by the impact of resolution of the issue on the Dresden-2 fault trees or events and the dominance or nondominance of accidents containing those faults or events. The impacts are developed from the SEP Branch evaluations of the issues and the Dresden-2 fault trees. The Dresden-2 fault trees were developed, in turn, from the IREP Millstone-1 fault trees and Dresden-2 plant information. The "dominance" of the Dresden-2 fault trees and events is determined from those fault trees, experience in performing the Millstone-1 IREP PRA, and the results of PRAs of other BWRs. The resulting classifications are given in Table 1.

A discussion of each issue and its classification is given in the Executive Summary of this report. The next section provides a brief overview of the results of this study.

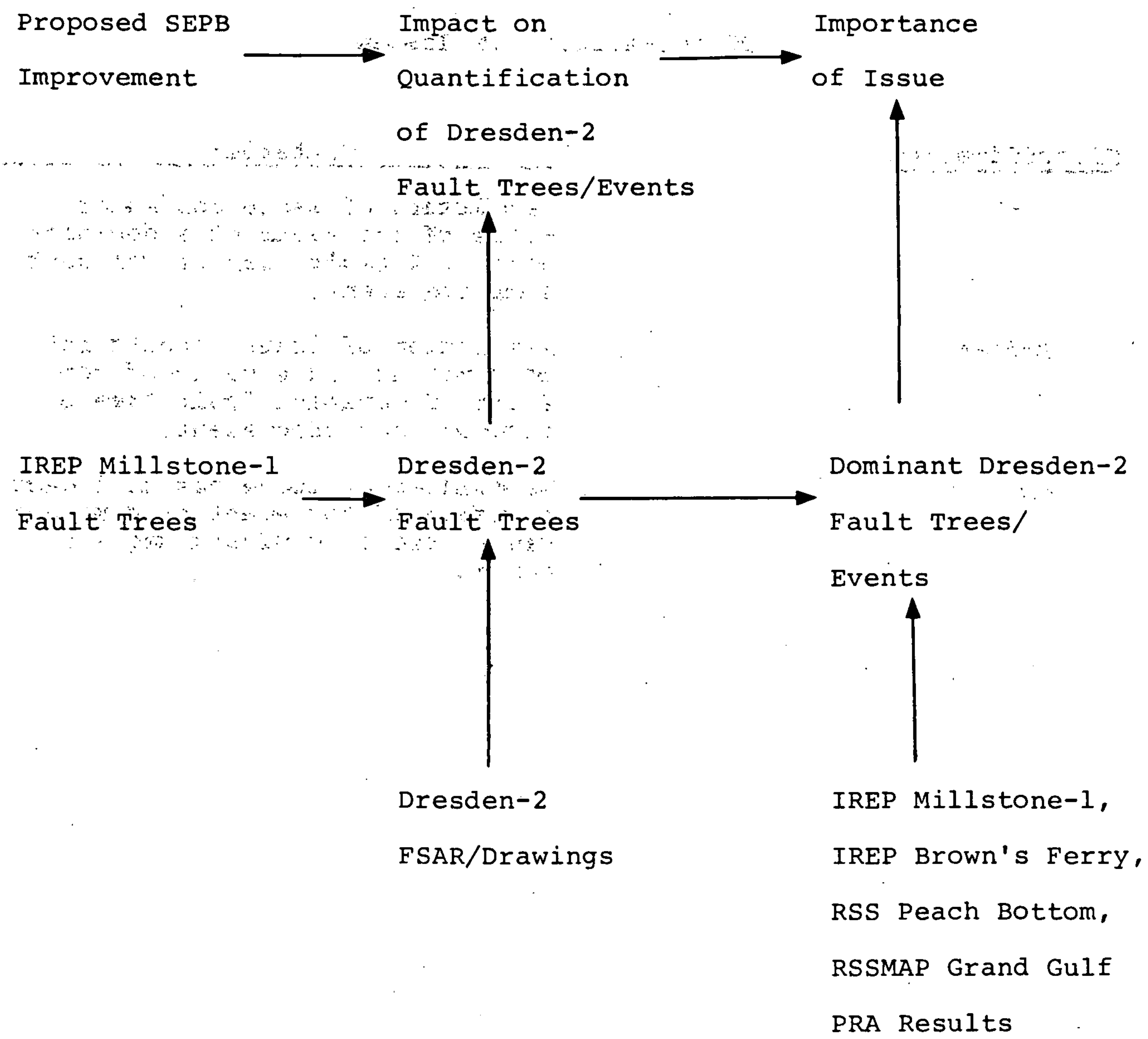

Figure 1. Study Methodology

TABLE 1

Classification of Issues

Classification	Criterion
High	Resolution of issue dominates value of top event of a dominant Dresden-2 fault tree or dominant sequence event.
Medium	Resolution of issue impacts but does not dominate value of top event of dominant fault tree or dominant sequence event.
Low	Resolution of issue has no impact on value of top event of dominant fault tree or dominant sequence event.

III. Results

There were 33 issues identified by the Systematic Evaluation Program Branch for the Dresden-2 nuclear power plant. Of these, 13 were outside the scope of our analysis, and 20 were at least partially within our scope. Table 2 gives those issues we did not analyze and Table 3 gives those issues we did analyze.

Each issue was analyzed for classification by the criteria described in the previous section of this report. That is, we assessed whether resolution of the issue would affect the Dresden-2 fault trees, quantified the effect, examined the fault trees to determine the resulting change in the top event(s), and reviewed other BWR PRAs, especially the IREP Millstone-1 study, to determine whether the affected fault trees would be part of dominant accident sequences.

Table 4 presents the results of our analysis. For each issue the following information is given: the system or accident event that the issue potentially impacts, the change in unavailability due to resolution of the issue and the component or system for which this was calculated, whether the fault tree(s) or event(s) affected would appear in any dominant accident sequences, whether the issue affects the top event of the fault tree(s)/event(s), and, based on applying the criteria of Section II to all of the above results, the resulting classification of the issues. Table 5 gives a list of the classifications of the issues as high, medium, or low importance to

TABLE 2

Issues Not Evaluated (13)

II-1.C	Potential Hazards Due to Nearby Transportation, Institutional, Industrial, and Military Facilities
II-3.B	Flooding Potential and Protection Requirements
II-3.B.1	Capability of Operating Plants to Cope With Design Basis Flooding Conditions
II-3.C	Safety-Related Water Supply (Ultimate Heat Sink)
III-1	Classification of Structures, Systems, and Components (Seismic and Quality)
III-2	Wind and Tornado Loadings
III-3.C	In service Inspection of Water Control Structures
III-4.A	Tornado Missiles
III-4.B	Turbine Missiles
III-5.A	Effects of Pipe Break on Structures, Systems, and Components Inside Containment
III-6	Seismic Design Considerations
III-7.B	Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria
V-6	Reactor Vessel Integrity

TABLE 3

Issues Evaluated (20)

III-5.B	Pipe Break Outside Containment
III-8.A	Loose-Parts Monitoring and Core Barrel Vibration Monitoring
III-10.A	Thermal Overload Protection for Motors of Motor-Operated Valves
V-5	Reactor Coolant Pressure Boundary (RCPB) Leakage Detection
V-11.A	Requirements for Isolation of High and Low Pressure Systems
V-11.B	RHR Interlock Requirements
VI-4	Containment Isolation System
VI-6	Containment Leak Testing
VI-7.C.1	Appendix K--Electrical Instrumentation and Control (EIC) Re-Reviews
VI-10.A	Testing of Reactor Trip System and Engineered Safety Features, Including Response Time Testing
VI-10.B	Shared Engineered Safety Features, On-Site Emergency Power and Service Systems for Multiple Unit Facilities.
VII-1.A	Isolation of Reactor Protection System From Nonsafety Systems, Including Qualification of Isolation Devices
VII-3	Systems Required for Safe Shutdown
VIII-2	On Site Emergency Power Systems--Diesel Generator
VIII-3.A	Station Battery Capacity Test Requirements
VIII-3.B	DC Power System Bus Voltage Monitoring and Annunciation
IX-5	Ventilation Systems
XV-1	Decrease in Feedwater Temperature, Increase in Feedwater Flow, and Increase in Steam Flow.
XV-16	Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment
XV-18	Radiological Consequences of a Main Steam Line Failure Outside Containment

TABLE 4

Results of Analysis

Issue	System/Component	Change in Unavailability q_{new}/q_{old}	Appears in Dominant Fault Tree/Event	Affects Top Event	Importance
III-5.B	Pipe break outside containment	---	No	No	Low
III-8.A	Transients	1.0 (transient frequency)	Yes	No	Low
III-10.A	Valves in all ECCS	0.86 (1 valve)	Yes	Yes	Medium
V-5	Small LOCA	1.0 (LOCA frequency)	No	No	Low
V-11.A	RWCU LOCA	1.2×10^{-6}	No	---	Low*
V-11.B	Shutdown Cooling	0.85 (shutdown cooling)	Yes	Yes	Medium
VI-4	Containment integrity	---	No	---	Low
VI-6	Containment integrity	---	No	---	Low
VI-7.C.1	AC and DC power	1.0 (AC or DC)	Yes	No	Low
VI-10.A	Reactor Trip System, Engineered Safety Features	1.0 (RTS)	Yes	No	Low
VI-10.B	AC and DC power	1.0 (DC power)	Yes	No	Low

*If pressure relief valve sufficiently sized. High importance if not sufficiently sized.

TABLE 4 (cont'd)

Issue	System/Component	Change in Unavailability q_{new}/q_{old}	Appears in Dominant Fault Tree/Event	Affects Top Event	Importance
VII-1.A	Reactor Trip System	1.0 (RTS)	Yes	No	Low
VII-3	Cooldown procedures	---	No	No	Low
VIII-2	AC power	0.98 (1 Diesel)	Yes	No	Low
VIII-3.A	DC power	6.5×10^{-2}** (1 battery)	Yes	Yes	High
VIII-3.B	DC Power	0.19 (1 bus)	Yes	Yes	High
IX-5	Ventilation	---	No	No	Low
XV-1	Power Conversion System	1.0	Yes	No	Low
XV-16	Offsite doses	---	No	---	Low
XV-18	Offsite doses	---	No	---	Low

**If present battery testing is totally ineffective.

TABLE 5

Classification of Issues
(Importance to Risk)

High		
	VIII-3.A	Battery Testing*
	VIII-3.B	DC Bus Instrumentation
Medium		
	III-10.A	Thermal Overload Trips
	V-11.B	Shutdown Cooling Interlock
Low		
	III-5.B	Pipe Break Outside Containment
	III-8.A	Loose Parts
	V-5	RCPB Leakage Detection
	V-11.A	RWCU LOCA**
	VI-4	Containment Penetrations
	VI-6	Containment Leak Testing
	VI-7.C.1	Electrical Distribution
	VI-10.A	Response Time Testing
	VI-10.B	DC Buses in Parallel
	VII-1.A	RPS Isolation
	VII-3	Shutdown Procedures
	VIII-2	Diesel Trips
	IX-5	Ventilation Systems
	XV-1	Turbine Bypass
	XV-16	Radiological Consequences
	XV-18	of Non-Core-Melt Events

*High importance if present battery testing is totally ineffective.

**Low importance if pressure relief valve sufficiently sized. High importance if not adequately sized.

risk. A discussion of the classification of each issue is given in the Executive Summary of this report.

IV. Analysis

Following are the analyses of the issues which were performed to assess their importances to risk.

III-5.B Pipe Break Outside Containment

1. NRC Evaluation

There are four areas where insufficient information is available to evaluate the degree to which the licensee meets current acceptance criteria:

1. The design of containment penetration piping between the containment and outboard isolation valve in the main steam lines, isolation condenser steam and condensate lines, and the reactor water cleanup inlet line.

2. The design of containment penetration piping outside of the outboard isolation valve in the isolation condenser steam line and the reactor water cleanup inlet line. The concern is that pipe failures could result in failure of the outboard isolation valve control and power systems.

3. The consequences, i.e., effects on the engineered safety features, of a main steam line break on the mezzanine floor of the turbine building.

4. The consequences, i.e., effects on the AC power system, of a reactor building closed cooling water leak (spray).

2. NRC Recommendations

The recommendations for each of these items are:

1. The licensee should demonstrate that a pipe break in the areas mentioned in Item 1 of Section 1 above could not occur or that the failure would not result in an unacceptably high risk.

2. The licensee should demonstrate that the stresses involved would not affect the outboard isolation valve.

3. The licensee should demonstrate that this break would not result in the disabling of emergency safeguards systems.

4. The licensee should demonstrate that this leak would not result in the disabling of the AC power system.

3. Systems Affected

The systems affected by the items examined in this analysis are: the isolation condenser system, the main steam system, and the reactor water cleanup system.

4. Comments

The effects of breaks on other systems will not be covered by this analysis. Whether the breaks postulated in Items 3 and 4 would affect emergency systems is beyond the scope of this analysis. A calculation can be provided for the probabilities associated with the unisolatable leaks produced by the breaks postulated in Items 1 and 2 and the failure of the inboard isolation valve.

5. Analysis

For the pipe breaks between the containment penetration and the outboard isolation valve the only additional failure required for an unisolatable break is a failure of the inboard isolation valve. Failure of the isolation valve is dominated by two events: failure of the valve to close on demand and failure of the valve circuit breaker to close on demand. Figures 1-4 show the piping of interest: the isolation condenser steam and condensate lines, the reactor water cleanup line and the main steam lines. Table 1 lists the data used in the analysis.

The main steam lines, isolation condenser steam line, and isolation condenser condensate line all contain one pipe segment between the isolation valve and the containment penetration. For each of these lines the frequency of an unisolatable break between the valve and the penetration is approximately

$$F = F(\text{pipe rupture}) \times [P(\text{MOV failure}) + P(\text{circuit breaker failure})]$$
$$= 8.7 \times 10^{-7}/\text{Ryr} \times (1 \times 10^{-3} + 1 \times 10^{-3})$$
$$= 1.7 \times 10^{-9}/\text{Ryr}.$$

In the reactor water cleanup system there are 13 pipe segments (including the two pipe reductions) between the isolation valve and the containment penetration. This includes the pipe segments leading to valve 2-1201-3. The frequency of an unisolatable break is:

$$F = F(\text{pipe rupture}) \times (\text{no. pipe segments}) \times [P(\text{MOV failure}) + P(\text{breaker failure})]$$
$$= 8.7 \times 10^{-7}/\text{Ryr} \times 13 \times (1 \times 10^{-3} + 1 \times 10^{-3})$$
$$= 2.2 \times 10^{-8}/\text{Ryr}.$$

To evaluate the second item in the NRC evaluation two assumptions are made. First, for the break to affect the operation of the isolation valve it must occur between the outboard isolation valve and the first pipe restraint. Second, any break that does occur in this part of the pipe is assumed to cause the isolation valve to fail. This produces the same situation as analyzed above. The inboard isolation valve must operate to isolate the break.

The two pipe segments of interest are on the isolation condenser steam line and the reactor water cleanup inlet line. There is only one pipe section between the isolation condenser steam line isolation valve and the first pipe restraint. Therefore the frequency of an unisolatable break is approximately

$$F = F(\text{pipe break}) \times [P(\text{inboard isolation valve fails to close}) + P(\text{inboard isolation valve circuit breaker failure})]$$

$$= 8.7 \times 10^{-7}/\text{Ryr} \times (1 \times 10^{-3} + 1 \times 10^{-3})$$

$$= 1.7 \times 10^{-9}/\text{Ryr}$$

The pipe section of interest in the reactor water cleanup system inlet line contains two pipe segments, including a pipe reduction and a check valve. The frequency of an unisolatable break in this section of pipe is:

$$F = [F(\text{pipe rupture}) \times (\#\ \text{pipe segments}) + F(\text{check valve rupture})] \times [P(\text{inboard isolation valve failure}) + P(\text{inboard isolation valve circuit breaker failure})]$$

$$= (8.7 \times 10^{-7}/\text{Ryr}\ (2) + 8.7 \times 10^{-5}/\text{Ryr})\ (1 \times 10^{-3} + 1 \times 10^{-3})$$

$$= 1.8 \times 10^{-7}/\text{Ryr}.$$

6. Conclusions

None of the unisolatable breaks postulated have an expected frequency of more than 2×10^{-7}/Ryr. When compared to the expected frequency of accident sequences with core melt and release probabilities on the order of 10^{-5}/Ryr the risk of these events is negligible. That is, even if we assume a core melt with release after any of these pipe breaks with no further independent failures, the effect on risk due to eliminating these breaks is unimportant. Note that this conclusion also applies to the high energy pipe break aspects of this issue since we do not specify any mechanism to cause the assumed core melt.

TABLE 1

Data Summary

Fault	Failure Rate	Exposure Time	Prob./Freq./ Unavailability
Pipe rupture (≥3-in.-diameter pipe)	1×10^{-10}/hr[1]	1 yr	8.7×10^{-7}/Ryr
MOV fails to close	1×10^{-3}/d		1×10^{-3}
Circuit breaker fails to close	1×10^{-3}/d		1×10^{-3}
Check valve--severe rupture	1×10^{-8}/hr	1 yr	8.7×10^{-5}/Ryr

[1]Failure rate is per section of pipe--a section is the pipe between two welds.

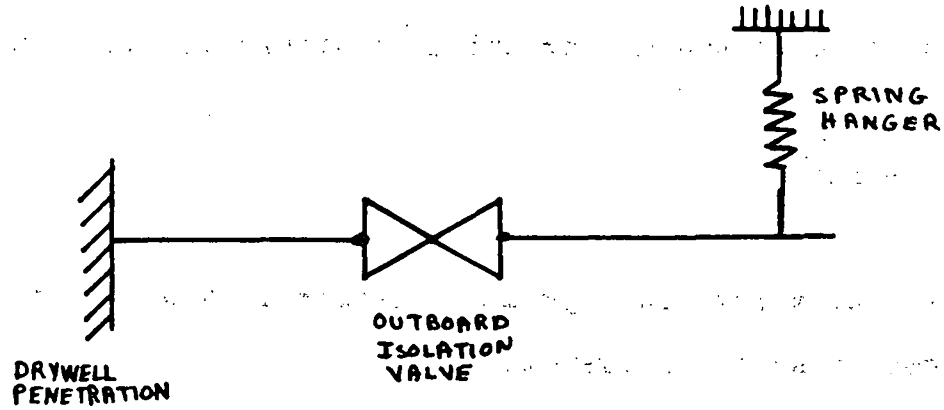

Figure 1. Isolation Condenser Steam Line

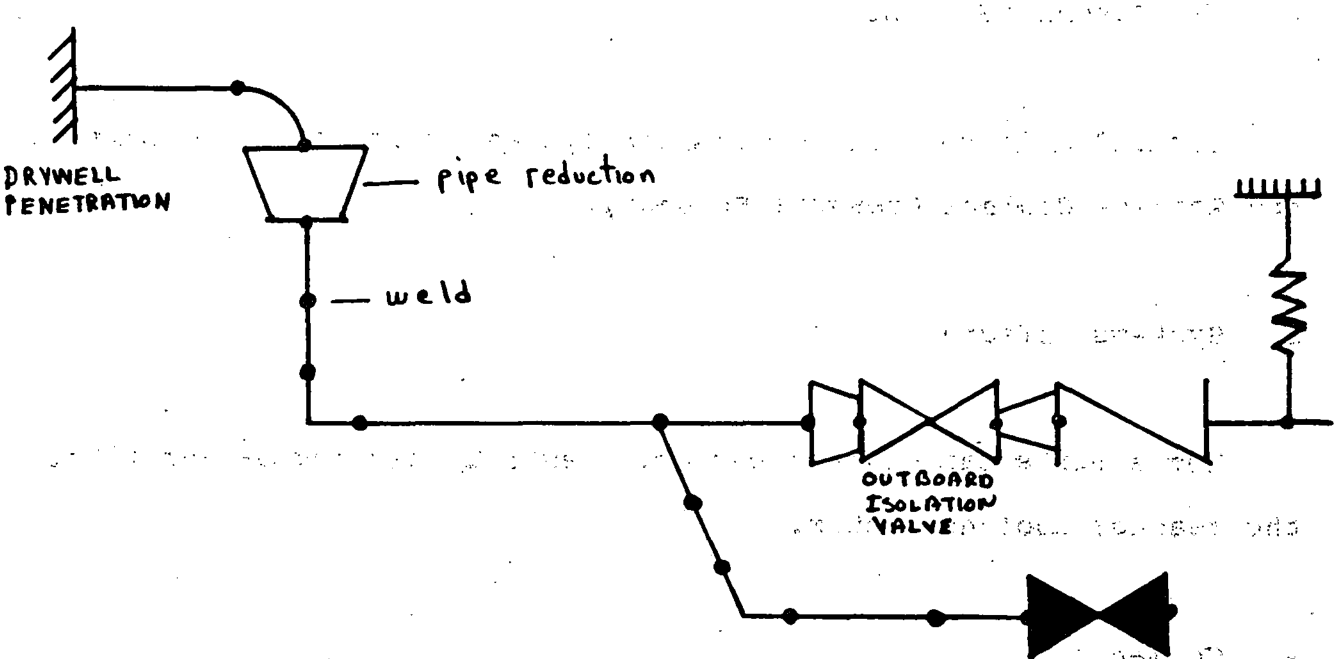

Figure 2. Reactor Water Cleanup Suction 2-1201-3

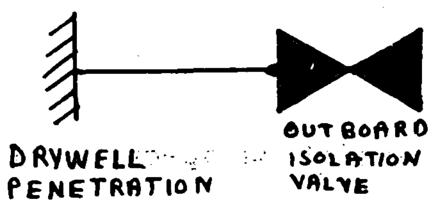

Figure 3. Isolation Condenser Condensate Line

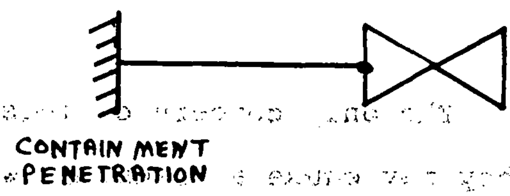

Figure 4. Main Steam Line (Typical of 4)

III-8.A Loose-Parts Monitoring and Core Barrel Vibration Monitoring

1. NRC Evaluation

A loose-parts monitoring system as required by Regulatory Guide 1.133 does not exist at Dresden-2.

2. NRC Recommendations

Install a loose-parts monitoring system to detect loose parts in the Reactor Coolant Pressure Boundary.

3. Systems Affected

Loose parts can cause transient events by causing damage within the reactor coolant system.

4. Comments

None.

5. Analysis

The only concern of loose parts from a risk perspective is that they may cause a transient which challenges the plant and its safety systems. There is ample data on transients to show that this effect is negligible. That is, because the historical transient rate is so high, several per reactor-year, and the contribution to this frequency by loose parts has been negligible, eliminating loose-parts-induced transients will have no effect on the transient frequency or risk.

In the USNRC Memorandum from L. S. Rubenstein to S. Hanauer, Director, DST, and D. Eisenhut, Director, DL, May 6, 1982, on the Loose Parts Monitoring Program, a history of loose parts effects to 1977 is given. There were forty-six "events" (loose parts), of which 23 were discovered by routine surveillance and 15 caused damage or malfunction. None of these "events" were transients requiring plant shutdown. Even if we use the approximate value of 20 transients caused by loose parts, the loose-parts induced transient rate is about 0.1/Ryr. The transient rate due to other causes is 7-10/Ryr.

6. Conclusions

Eliminating loose-parts-induced transients by installing a loose-parts monitoring system would have no effect on risk.

III-10.A Thermal Overload Protection for Motors of Motor-Operated Valves

1. NRC Evaluation

Thermal overload protection for MOVs should be bypassed, under accident conditions, by the ECCS signal or the trip setpoints should be high enough to prevent spurious trips due to design inaccuracies, trip setpoint drift, or ambient temperature variations. At Dresden-2, there are no thermal overload protection devices that are bypassed.

2. NRC Recommendation

Design Modifications should be provided to override the thermal overload protection with an ECCS signal. Additionally, the NRC recommends that torque switches should be bypassed with a limit switch during automatic valve actuation.

3. Systems Affected

Since no thermal overloads are bypassed at Dresden-2, all systems that are part of the ECCS are affected.

4. Comments

Spurious trips of the thermal overload protection for an MOV will add to the MOV unavailability. The spurious trip signal will prevent the MOV from opening even though there is nothing wrong with the valve itself. However, by bypassing the thermal overload protection the danger of damaging the valve increases, and this reduces the possibility of recovering the operability of the valve.

5. Analysis

The failure rate for an MOV can be found in Appendix III of the Reactor Safety Study (WASH-1400). This unavailability, per demand, is

$$\lambda_D(\text{MOV}) = 1 \times 10^{-3}.$$

This is based on monthly testing of the valve operability.

Since thermal overloads are not bypassed during value surveillance, this failure rate includes the contribution of the thermal overload protection device failures.

The failure rate of the thermal overload can be found in Section 1 of the Nonelectronic Parts Reliability Data (NPRD-2). The failure rate for a thermal relay is

$$\lambda_S = 4 \times 10^{-7}/\text{hr}.$$

This operating failure frequency must be compared to the demand unavailability of the MOV. To convert the standby failure rate to a demand failure rate, the following equation is used

$$\lambda_D = \frac{1}{2}\lambda_S t$$

where t is the time between tests. Since the WASH-1400 data is based on a monthly test interval to compare the failure rates, a test interval of one month is used for the thermal relay failure data. This yields a thermal switch demand failure rate of 1.4×10^{-4}. By bypassing the thermal overload this contribution to valve failure would be eliminated and the failure rate of an MOV with a bypassed thermal overload device can be expected to be 8.6×10^{-4} per demand. This is approximately a 14 percent decrease in the MOV unavailability.

To evaluate the effect this reduction in the MOV failure rate would have on the risk due to core melt, the dominant sequences that lead to core melt must be examined. The sequences evaluated are those from the Millstone-1 IREP study. The Millstone-1 and Dresden-2 ECCS are very similar, the primary difference being the presence of an HPIS in Dresden-2 as opposed to the FWCI system in Millstone-1. The data used for the Millstone IREP study represents the MOVS as they are now in Dresden. The reduction in the MOV failure rate affects only a few of the dominant sequences. The effect is to reduce these sequence frequencies, but by less than 5 percent for any given sequence for Millstone. Modified fault trees and sequences used to represent Dresden showed that this issue would have a similar effect on plant risk at Dresden-2 as it did for Millstone-1.

Although bypassing the thermal overloads for the MOVs has an effect on the component failure rate, the effect on core melt risk is minimal.

The issue of overriding the torque switches with limit switches during automatic valve actuation can be analyzed by comparing the failure rates of both devices. From WASH-1400 the failure rate of a torque switch is 1×10^{-4} per demand and the failure rate of a limit switch is 3×10^{-4} per demand. Overriding the torque switch with a limit switch does not replace the torque switch with a more reliable device. Therefore it cannot be expected that overriding the torque switch will improve the availability of the MOV.

6. Conclusion

The failures of motor-operated valves do not significantly affect risk. The reduction in the failure rates for the MOVs achieved by bypassing the thermal overloads would not have any significant effect on risk due to core melt.

V-5 Reactor Coolant Pressure Boundary (RCPB) Leakage Detection

1. NRC Evaluation

The Dresden-2 reactor coolant pressure boundary (RCPB) leakage detection system should be able to detect a 1-gallon/minute (1-gpm) leak in 1 hour. It has not been demonstrated that the equipment has this sensitivity.

2. NRC Recommendations

Install equipment or ensure that the present equipment can detect a 1-gpm leak in 1 hour.

3. Systems Affected

The system affected is the reactor coolant system. This issue potentially affects risk through the frequency of small LOCAs.

4. Comments

The NRC hypothesis is that early leak detection may allow operator action to isolate the leak or shut down (depressurize) the plant thereby preventing the leak from becoming a LOCA. This is the "leak-before-break" issue for pipes.

By technical specifications, the plant must shut down upon discovering a 5-gpm unidentified leakage. The maximum cooldown rate is 100°F/hr.

There are several unknowns associated with assessing the impact of leak detection time on preventing LOCAs: the mean time it would take a leak to grow to LOCA proportions, the fraction of leaks which, in fact, become LOCAs, and the ability of the operators to prevent a LOCA upon discovering a small leak.

Dresden-2 presently has at least one system for detecting leakage which approaches the required 1 gpm in 1 hour sensitivity. The sump pump flow integrator can detect a 1/2-gpm (200-gallon) leakage when the sumps are pumped every 8 hours. Thus at present, Dresden-2 can detect 1-gpm leakage in about 8 hours by this method.

A parallel concern is that the leak (pipe crack) may not grow naturally, but if the plant experienced a transient before the leak was detected and the plant shut down, the transient could cause the crack to become a break, i.e., a LOCA. We do not analyze this concern here because of lack of data on these kinds of leaks. In addition, we do not consider the high energy pipe break (system interactions) aspects of the breaks because PRAs do not assume any additional failures caused by LOCAs.

5. Analysis

A fault tree for small LOCA frequency, incorporating leakage detection, is given in Figure V-5-1. The impact of this issue is to change the quantification of the leakage detection probability. The assumptions we make to maximize the calculated impact of this issue are that all pipe break LOCAs as calculated in WASH-1400 begin as leaks which can be prevented from becoming LOCAs, and the change in leakage detection probability is maximized.

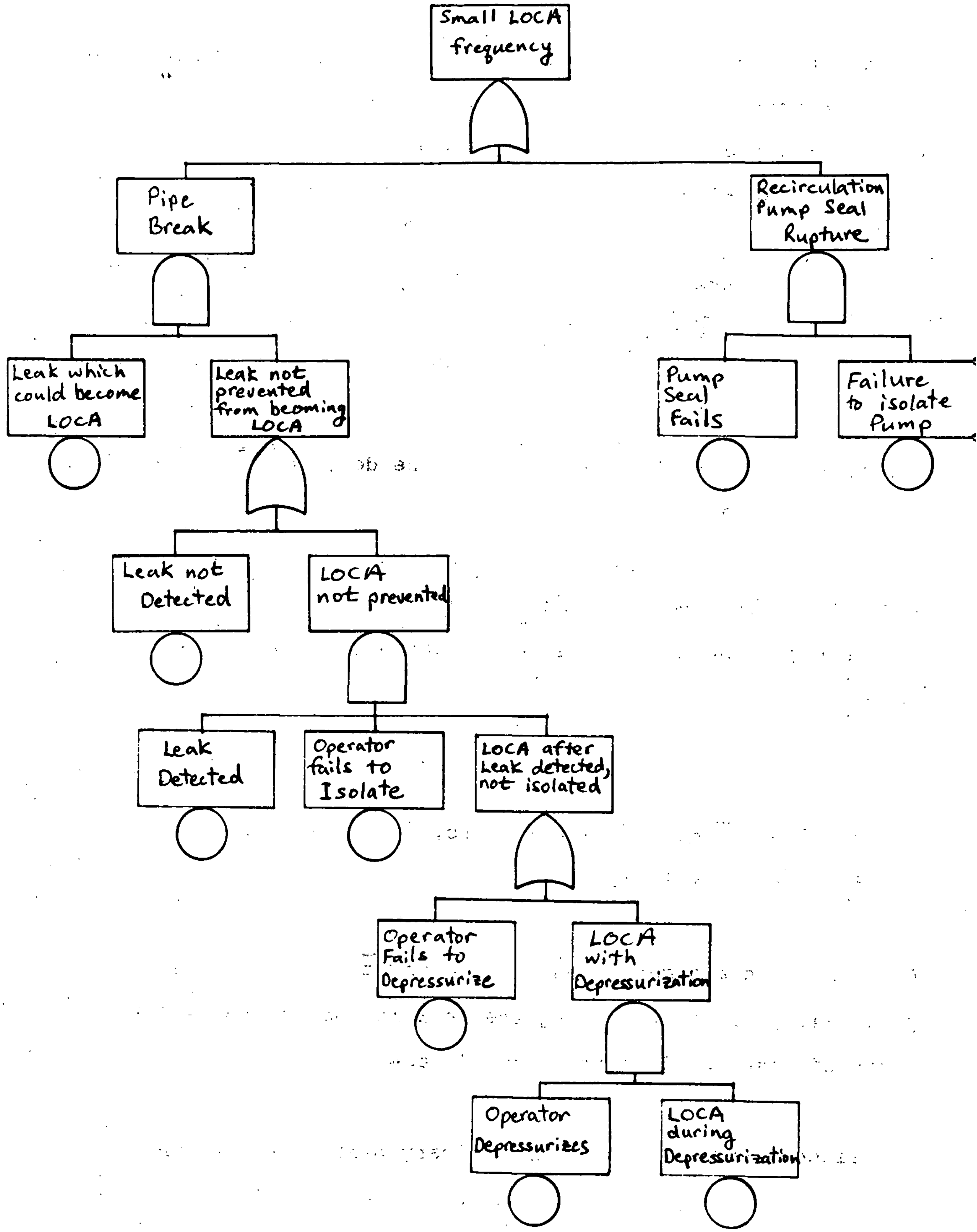
Small LOCA frequency
Pipe Break
Recirculation Pump Seal Rupture
Leak which could become LOCA
Leak not prevented from beoming LOCA
Pump Seal Fails
Failure to isolate Pump
Leak not Detected
LOCA not prevented
Leak Detected
Operator fails to Isolate
LOCA after Leak detected, not isolated
Operator Fails to Depressurize
LOCA with Depressurization
Operator Depressurizes
LOCA during Depressurization

Figure V-5-1.

The change in leakage detection probability is maximized by assuming that the detection time is presently 8 hours and by considering the mean time it would take for a leak to become a LOCA, $\bar{t}$. This mean time is unknown. There are three possibilities:

1. $\bar{t} > 8$ hrs.
2. 1 hr. $< \bar{t} < 8$ hrs.
3. $\bar{t} < 1$ hr.

The probability that a leak will not be detected is, approximately,

$$P_{ND} = \begin{cases} 0, & \bar{t} > t_d \\ 1 - \bar{t}/t_d, & \bar{t} < t_d, \end{cases}$$

where t_d is the detection period (here being changed from 8 hours to 1 hour). The difference in detection probability is, in each case,

1. 0 , $\bar{t} > 8$ hrs.
2. $1 - \bar{t}/8$, 1 hr $< \bar{t} < 8$ hrs.
3. $7\bar{t}/8$, $\bar{t} < 1$ hr.

Cases b and c are maximized by choosing $\bar{t} = 1$ hour. Thus the probability of not detecting the leak is assumed to be 7/8 in the "before" case and 0 in the "after" case.

Since virtually all of the primary system (except the recirculation pumps) is unisolatable, the probability that the operator cannot isolate the leak is 1.0.

The recirculation pump seal failure rate is taken from the data in the December 9, 1980, NRC Memorandum for E. Adensam, Acting Branch Chief, Operating Experience Evaluation Branch from R. Riggs, Operating Experience Evaluation Branch. There has been one pump seal failure in 53 pump years at jet pump BWRs. Thus the failure rate is about 1.9×10^{-2}/pump-yr. There are two recirculation pumps at Dresden-2, so the overall failure rate is 3.8×10^{-2}/yr. These LOCAs would be isolatable by the recirculation pump isolation valves. We assume that the dominant cause of failure to isolate is operator error (probability $\sim 3 \times 10^{-3}$) as the valve failure probability is much less ($\sim 1 \times 10^{-4}$).

If the leak is detected, there may remain some time to prevent the leak from becoming a LOCA. This is the whole point of requiring this leakage detection capability. However, if we choose the time for the leak to become a break to be 1 hour, then in both our "before" and "after" cases, by the time the leakage detection system integrates the leakage for 1 hour, or 8 hours, there is no time left to prevent the LOCA. Thus the effect of leakage detection is negligible.

We also analyzed the situation where the time for the leak to become a break is longer, chosen to be 2.5 hours. This allows time for action to depressurize the plant and decrease the likelihood that the leak becomes a LOCA. In the "before" case, the integration time is still longer than the time for the leak to become a LOCA, so preventative actions are still ruled out. In the "after" case, there remains 1.5 hours to depressurize after the leak is detected. The stress on the leaking pipe is proportional to the pressure, and if we assume that the probability of the crack (leak) propagating is proportional to the stress, we can estimate the probability of a LOCA during depressurization as the ratio of the pressure after 1.5 hours of depressurization to the operating pressure (since we assume that the probability of LOCA is 1.0 without depressurization). At the maximum cooldown rate of 100°/hr on the saturation curve after 1.5 hours from the operating pressure of 1020 psig, the pressure will be 240 psig. Thus the probability of LOCA during depressurization is 240/1020 = 0.24.

The above argument assumes that the operator will depressurize upon discovering a 1-gpm unidentified leakage. However, the Technical Specifications do not require plant shutdown until a leakage of 5-gpm unidentified leakage is reached. Discussions with Dresden-2 personnel indicate that if, for example, 500 gallons of leakage were discovered in the sumps at the shift sump pumping operation (every 8 hours), which represents a 1-gpm unidentified leakage, the operators would enter containment to attempt to identify the leakage, but would most probably not shut down the plant until the 5 gpm Technical Specification limit was reached. Thus we assess the probability that the operator would actually shut down at 1 gpm as very small, $\sim 1.0 \times 10^{-2}$.

Using these data, the fault tree in Figure V-5-1 was quantified. If the time for the leak to become a break is 1 hour, then there is no change in the small LOCA frequency of 1.1 x 10^{-3}/yr from decreasing the leakage detection time from 8 hours to 1 hour. If the time for the leak to become a break is longer, say 2.5 hours, then the decreased leakage detection time will also not decrease the small LOCA frequency from 1.1 x 10^{-3}/yr. This is because it is unlikely that the operator will shut down the plant when a 1-gpm leakage is detected. (We increased the probability that the operator will depressurize from 1.0 x 10^{-2} to 0.1 and this change had no significant effect on the LOCA frequency.)

If the plant were required to shut down at 1 gpm leakage, then the change in LOCA frequency would be from 1.1 x 10^{-3}/yr at 8 hours detection time to 3.5 x 10^{-4}/yr at 1 hour detection time if the time for the leak to become a break is 2.5 hours.

6. Conclusion

Because Dresden-2 is not required to shut down, by Technical Specifications, until a 5-gpm unidentified leakage is discovered, increasing the sensitivity of leakage detection to detect a 1-gpm leak in 1 hour will have little or no effect on the small LOCA frequency.

TABLE V-5-1

Fault Tree Quantification Data

Event Name/Description	Unavailability or Frequency		
Pump seal fails	3.8×10^{-2}/yr		
Failure to isolate pump	3×10^{-3}		
Leak which would become LOCA	1×10^{-3}/yr		
Operator fails to isolate leak	1.0		
Leak not detected	$\bar{t}$ = 1 hr	0.875	before
		0	after
	$\bar{t}$ = 2.5 hr	0.688	before
		0	after
Leak detected	$\bar{t}$ = 1 hr	0.125	before
		1.0	after
	$\bar{t}$ = 2.5 hr	0.313	before
		1.0	after
LOCA during depressurization	$\bar{t}$ = 1 hr	1.0	before
		1.0	after
	$\bar{t}$ = 2.5 hr	1.0	before
		0.24	after
Operator depressurizes	1.0×10^{-2}		
Operator fails to depressurize	0.99		

V-11.A Requirements for Isolation of High- and Low-Pressure Systems

1. NRC Evaluation

The RWCU (Reactor Water Cleanup System) does not have independent interlocks on the suction valves and does not satisfy current criteria.

2. NRC Recommendation

Redundant high-pressure interlocks are to be installed on the RWCU suction valves.

3. Systems Affected

The only system affected is the RWCU through the probability of an interfacing system LOCA.

4. Comments

The ability of the pressure relief valve on the RWCU suction line to relieve any overpressure condition has not been adequately determined at this time. This analysis will be performed assuming that the pressure relief valve can fulfill its designed function.

5. Analysis

A fault tree for the failure of the RWCU system to isolate due to high pressure in the suction line is shown in Figure 1. The two MOVs used for isolation and the pressure control valve all operate on signals from one pressure sensor. Other valves are present but have no provisions for automatic closure (and they are normally open). If the automatic isolation devices do not function properly the operator would be unable to prevent the loss of at least some coolant. For this reason no credit is taken for operator actions to isolate the high-pressure leak.

The data used, shown in Table 1, is based on annual surveillance of the valves. (Dresden has a year-long fuel cycle and the assumption is made that the valves are tested during refueling outages.)

The frequency of a high-pressure leak in the RWCU system is primarily due to the combination of a pressure sensor failure and a failure of the pressure relief valve. The failure of the pressure sensor affects the pressure control valves and both MOVs. For this system the frequency of a high-pressure leak leading to a LOCA outside containment is 2.4E-7/Ryr. (Given that the isolation devices do not function it is assumed that a LOCA will occur.)

$$\begin{aligned}\text{Frequency} &= \text{F(sensor failure)} \times \text{P(relief valve failure)} + \\ &\quad \text{P(MOV fails)} \times \text{P(MOV fails)} \times \text{F(pressure control valve fails)} \times \text{P(relief valve failure)} \\ &= 2.4\text{E-}3/\text{Ryr} \cdot 1\text{E-}4 + (1\text{E-}2)(1\text{E-}2)(7\text{E-}4/\text{Ryr})(1\text{E-}4) \\ &= 2.4\text{E-}7/\text{Ryr}\end{aligned}$$

If it is determined that the pressure relief valve is not sufficiently sized to prevent damage in the low-pressure portion of the RWCU system then the frequency of a LOCA outside containment due to this system increases to 2.4E-3/Ryr (the frequency of the pressure sensor failure).

The addition of an independent pressure sensor to provide an isolation signal to one of the MOVs is modeled in the fault tree shown in Figure 2. This change yields the following frequency for a LOCA outside the containment:

Frequency = [P(MOV failure) + P(sensor failure)] x [F(sensor failure) + (P(MOV failure) x F(PCV failure)] x P(relief valve failure)

= [1E-2 + 1.2E-3][2.4E-3/Ryr + (1E-2)(7E-4/Ryr)](1E-4)
= (1.1E-2)(2.4E-3/Ryr)(1E-4)
= 2.8E-9/R-yr

Again, if it is determined that the pressure relief valve is insufficiently sized this value increases to 2.8E-5/Ryr.

The only other concern is that if the relief valve is sufficiently sized, and it passes flow due to the pressure sensor failure, then a large LOCA exists. This large LOCA frequency would be decreased from 2.4×10^{-3}/yr to 2.8×10^{-5}/yr due to resolution of this issue. Review of the Millstone-1 IREP PRA and the Dresden-2 fault trees showed that this change in large LOCA frequency would have no effect on risk (a large LOCA frequency of 2.4×10^{-3}/yr would not significantly contribute to the core melt frequency).

This is further complicated because the pressure relief valve does not return the coolant to the containment and a continued loss of coolant could eventually lead to a LOCA outside the containment through a different path than failure of the RWCU system piping. The following short analysis addresses this concern.

If the pressure relief valve opens, the coolant is released into the condenser hotwell. If the release should go unabated, it could lead to a LOCA outside the containment. To prevent this, the containment isolation system must isolate the reactor water clean up system. For this system, isolation is initiated by a low reactor water level which produces a signal closing two MOVs (MOV 1201-2 and MOV 1201-1).

The isolation signal is a dual channel signal, i.e., both channels must produce an isolation signal to close the MOVs. Each channel contains two sensors, one of which must function to produce an isolation signal in the channel. A simplified fault tree for the failure to isolate the reactor water cleanup system is shown in Figure 3. The data for this fault tree is shown in Table 2. This data uses the technical specification surveillance requirements for the test intervals that yield the exposure times (1/2 the test interval).

From this data and the fault tree, the failure to isolate the RWCU system on demand has a probability of 1.2E-4. When combined with the frequency of a sensor failure, causing the need for isolation, the frequency of an unisolated leak in the RWCU system is

$$(1.2E{-4})(2.4E{-3}) \approx 3E{-7}/Ryr.$$

This figure is still significantly lower than the expected frequency of a core melt and is therefore of little or no significance in evaluating the plant risk.

6. Conclusion

Provided that the pressure relief valve is sufficiently sized, the frequency of the LOCA due to failure of the isolation devices is extremely small. The pressure relief valve provides the redundancy that independent pressure signals would. The system as it is now, assuming the pressure relief valve is sufficiently sized, makes a smaller contribution to containment releases than a system with redundant interlocks and no pressure relief valve would.

TABLE 1

Data for RWCU System High Pressure Interlock

	$\lambda_H (hr^{-1})$	Exposure Time	Frequency/ Unavailability
MOV failure to close	2.7E-6	0.5 yr	1E-2
Pressure sensor does not function properly	2.7E-7	0.5 yr	1.2E-3
--frequency	2.7E-7		2.4E-3/Ryr
Pressure control valve fails open	2.7E-6		2.4E-2/Ryr
Pressure relief valve fails on demand	2.7E-8	0.5 yr	1E-4

Data is adapted from WASH-1400.

TABLE 2

RWCU System Isolation Failure Data

	λ_S (hr^{-1})	Exposure Time	Unavailability
MOV failure to close	2.7E-6	.5 yr	1E-2
Relay fails to deenergize	2.7E-7	.5 yr	1E-3
False Output from low water level sensor[1]			6E-4
a) sensor failure	2.7E-7	1000 hrs.	2.7E-4
b) relay failure to deenergize	2.7E-7	1000 hrs.	2.7E-4

[1] Consists of a relay failure or a sensor failure.

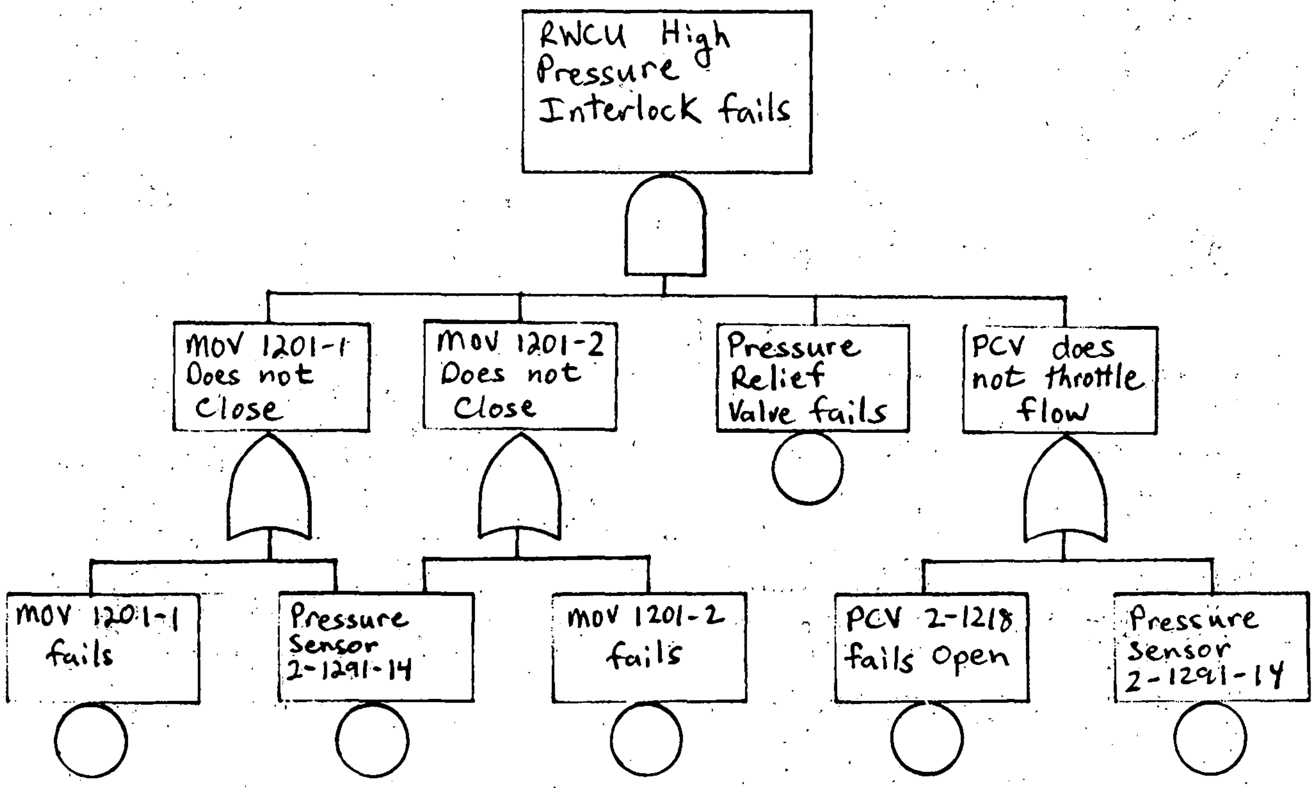

Figure 1. Dresden RWCU High-Pressure Interlock Failure Fault Tree

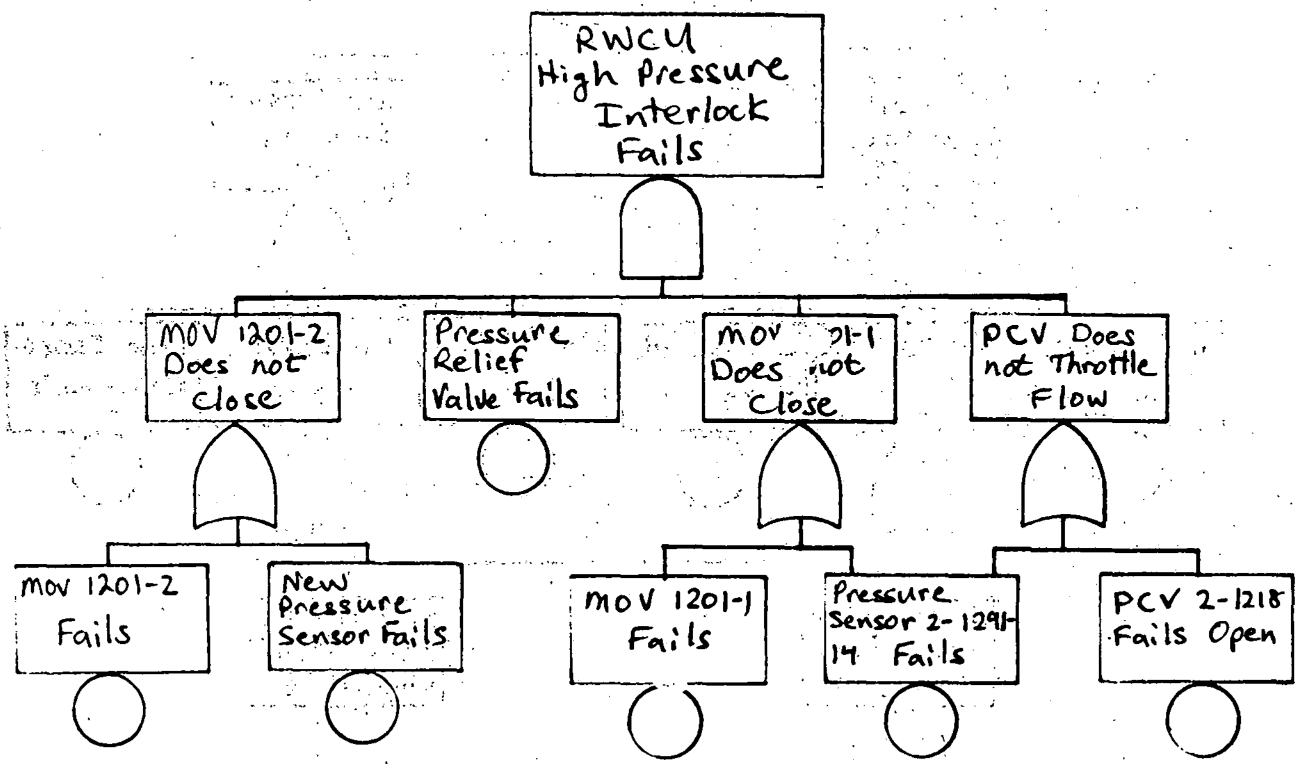

Figure 2. Modified Dresden RWCU System High-Pressure Interlock Failure Fault Tree

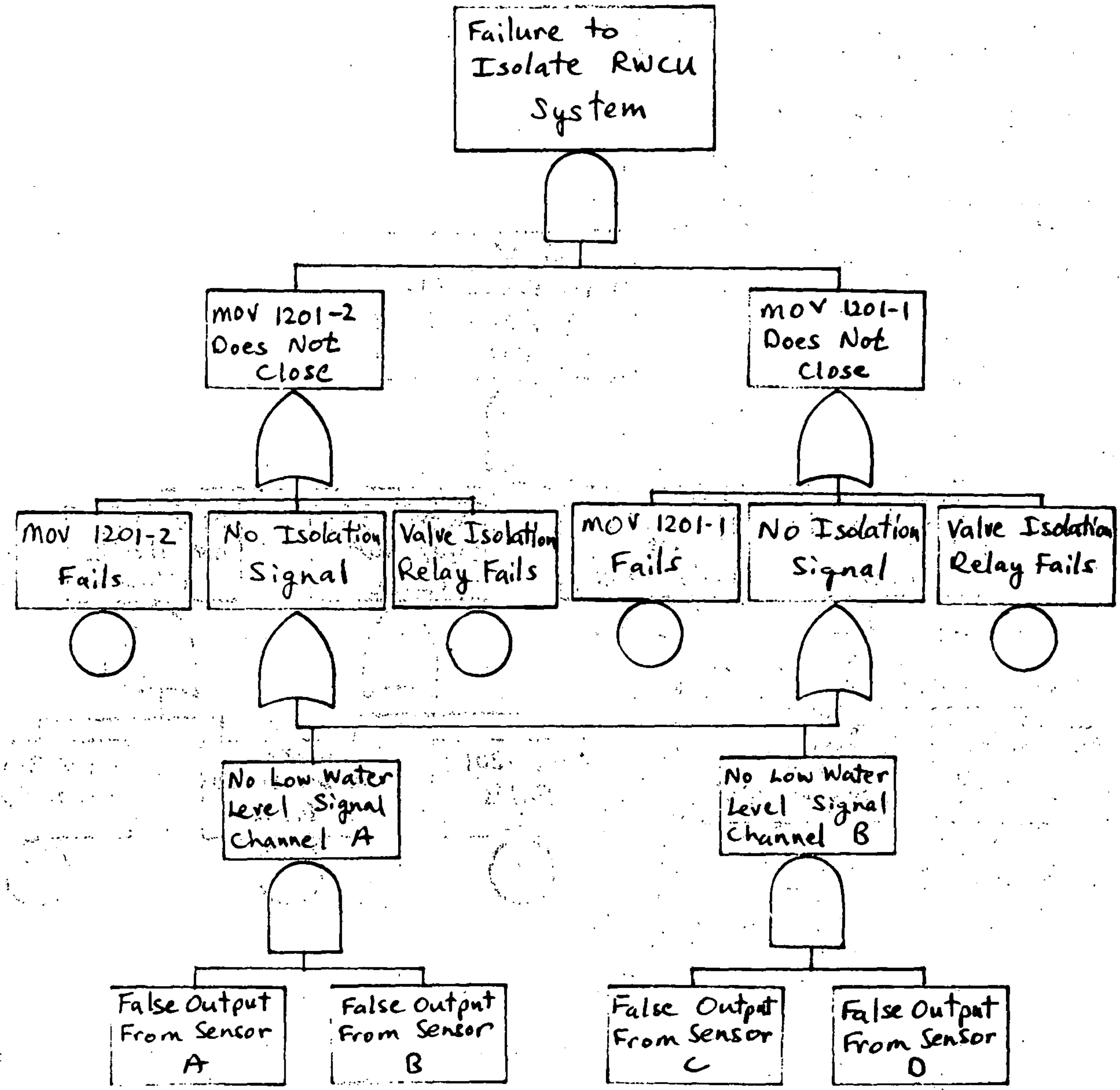

Figure 3. RWCU System Failure to Isolate Fault Tree.

V-11.B RHR Interlock Requirements

1. NRC Evaluation

The shutdown cooling system is designed for full reactor pressure, but less than full reactor temperature. Therefore, system interlocks are based on temperature requirements. Current licensing criteria for the interlocks are not met since there are no testing requirements.

2. NRC Recommendations

Require testing of the shutdown cooling interlocks.

3. Systems Affected

This issue affects the shutdown cooling system.

4. Comments

We assess this issue assuming that the thermal interlock has never been tested and will not be if not required. Also, we assume that if the shutdown cooling system is exposed to greater than design temperature, it will fail.

5. Analysis

The fault tree for failure of shutdown cooling incorporating the above assumptions is given in Figure V-11.B-1. The dominant causes of failure of shutdown cooling, from the IREP Millstone-1 and Dresden-2 fault trees, is failure of the pairs of inlet and outlet valves (each pair of which is powered by the same ac train). The data for the solution of this tree is given in Table V-11.B-1. The "before" case assumes that the temperature interlock is never tested and the "after" case assumes that the interlock is tested yearly. Resolution of this issue would decrease the overall unavailability of shutdown cooling from 4.7×10^{-3} to 4.0×10^{-3}, a decrease of about 15%.

6. Conclusion

Resolution of this issue could slightly affect the unavailability of shutdown cooling if exceeding the design temperature would always fail the system.

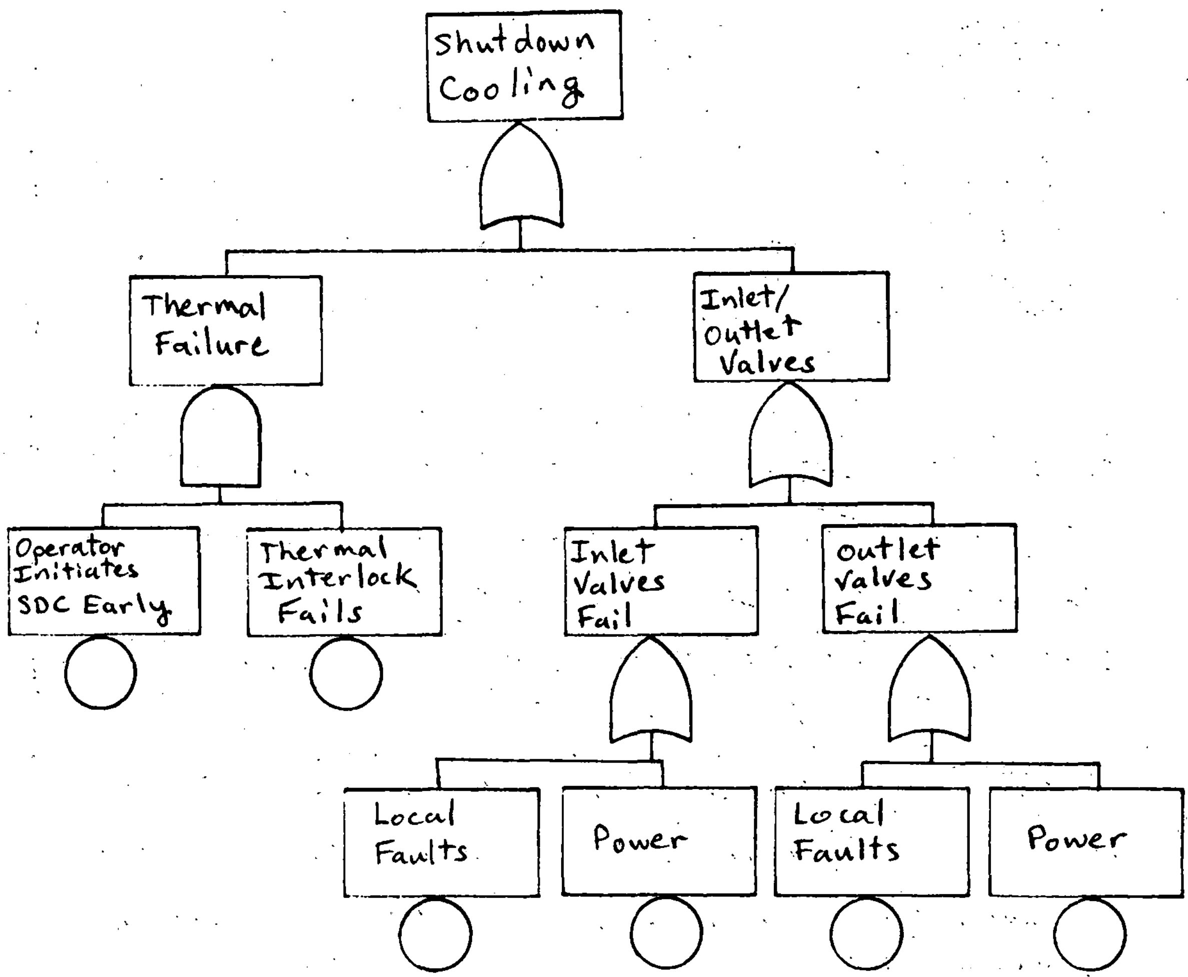

Figure V-11.B-1. Shutdown Cooling Fault Tree

Table V-11.B-1. Data Summary

Event	Sub-Event	Failure Rate	Exposure Time	Unavailability
Operator Initiates SDC Early				3×10^{-3}
Thermal Interlock Fails				
	Before Temperature Sensor	8.9×10^{-7}/hr	2.6×10^{5}hr	2.3×10^{-1}
	Relay Shorts Across NO contacts	1×10^{-8}/hr	2.6×10^{5}hr	2.6×10^{-3}
				2.3×10^{-1}
	After Temperature Sensor	8.9×10^{-7}/hr	8.8×10^{3} hr	7.8×10^{-3}
	Relay	1×10^{-8}/hr	8.8×10^{3} hr	8.8×10^{-5}
				7.9×10^{-3}
Local Valve Faults				
	Fails to Operate			1×10^{-3}
	Plug			1×10^{-4}
				1.1×10^{-3}
	2-valves		$(1.1 \times 10^{-3})^2 =$	1.2×10^{-6}
Power to Valve				
	Probability that transient was loss of offsite power and appropriate diesel generator fails		$(.2/10) \times 0.1 =$	2×10^{-3}

VI-4 Containment Isolation System

VI-6 Containment Leak Testing

1. NRC Evaluation

Both of these issues address the adequacy of containment integrity during accident conditions. Issues VI-4 identifies many containment penetrations which could have high probability of failure to isolate and Issue VI-6 identifies penetrations which have been requested to be exempt from leak testing requirements.

2. NRC Recommendations

Backfit the necessary hardware and testing to make containment penetrations conform to the GDCs and ensure containment integrity during accident conditions.

3. Systems Affected

These issues affect the nature of the release of radioactive material in an accident and thus the consequences of the accident.

4. Comments

We will not analyze each penetration here, but will show that the whole issue of containment isolation is not important to risk for Dresden-2.

5. Analysis

Because of the small size and low design pressure of the Dresden-2 containment (similar to Millstone-1), the pressure generated by steam and noncondensible gases during a core melt accident will most certainly fail the containment if another failure mechanism does not occur first. For example, in the Millstone-1 IREP PRA, all the accident sequences fail the containment by overpressure if a steam explosion does not occur.

The effect of changing the effectiveness of isolation of the penetrations is to shift the containment failure mode between leakage through the failed penetration and overpressure rupture (if the leakage would be great enough to prevent rupture). In the IREP Millstone-1, IREP Brown's Ferry, Reactor Safety Study Methodology Applications Program Grand Gulf, and Reactor Safety Study Peach Bottom PRAs of BWR plants, no dominant risk sequences involved isolation failure of the penetrations as the release mechanism. The overpressure failures for the Peach Bottom, Grand Gulf, Brown's

Ferry, and Millstone PRAs were BWR Release Categories 2 and 3 releases. Containment leakages (failure to isolate penetrations) were Release Category 4 releases in Grand Gulf, Millstone, Brown's Ferry, and Peach Bottom.

The smaller numbers are the greater consequence release categories. Thus improving the isolation could only decrease the containment leakage releases in Category 4 and increase the over-pressure releases in the higher consequences Categories 2 and 3, thus increasing risk.

6. Conclusion

Because the Dresden-2 containment will fail by overpressure in a core melt accident if no other failure occurs first, improving con-tainment isolation will not decrease the probability of release during such an accident or lower the consequences of the release.

VI-7.C.1 Appendix K--Electrical, Instrumentation and Control (EIC) Re-reviews

1. NRC Evaluation

Of the many items evaluated by the NRC for this issue the following items are the only ones where any actions have been recommended:

1. The connections between the buses supplying power to the DG2/3 control and instrumentation bus provide the potential for a single fault to affect the redundant 125-V dc power trains. This is possible primarily if the circuit breakers are not properly sized and coordinated.

2. The two 480-V switch gears 28 and 29 can be connected through the closure of breakers 2829 and 2929. This ties the two ac power trains together. There are no LCO requirements in regard to how long these breakers may be closed, and there is no provision for automatic opening of the breakers given a loss of offsite power.

3. There are no limiting conditions for plant operation that limit the amount of time that the instrumentation and control power for diesel generator 2/3 is connected to the Unit 3 125-V distribution panel, which receives its power from the Unit 2 Division II power source.

4. The 480-V ac switch gear 27 is powered from bus 24, with dc control power from Division I. Switch gear 27 is non-class 1E and the connection to a Class 1E bus is a deviation from review guidelines.

2. NRC Recommendations

The following recommendations are made for each of the above items:

1. Short-circuit analysis should be performed to demonstrate the adequacy of the ac and dc breakers associated with the diesel generator 2/3 control power.

2. LCO requirements should be added to the Technical Specifications prohibiting operation of the Class 1E systems in parallel while the plant is not shut down. Additionally, disconnect links should be opened in each circuit with a normally open circuit breaker, and the operating procedures should prohibit the use of breakers 2829 and 2929 during normal operation.

3. An LCO requirement limiting the time during which the instrument and control power for diesel generator 2/3 may be obtained from the Unit 3 125-Vdc distribution panel when the plant is not shut down should be implemented.

4. Short-circuit analysis should be performed on all circuit breakers between class 1E and non-class 1E buses.

3. Systems Affected

The electric power systems (ac and dc) are the systems affected.

4. Comments

Items 1 and 4 in Sections 1 and 2 will not be addressed in this analysis. Short-circuit analysis is used to determine the adequacy of the equipment installed to perform a particular function. In a PRA one of the assumptions made is that the equipment is of an adequate design to perform the function for which it was intended. (For example, a pump that is supposed to provide a required flow will provide that flow when it operates properly.)

The analysis of the remaining two items is based on discussions with plant personnel.

5. Analysis

The two ac buses, 28 and 29, are not normally connected during power operation. The only time the circuit breakers between the two 480-V switch gears are supposed to be closed is when, in response to a loss of offsite power, one of the diesel generators fails to start or run. In this situation, when it is required that loads on the dead ac train receive power, the two buses are connected. At this point Dresden-2 would not be in a normal power mode of operation.

This condition also applies to the DG 2/3 instrumentation and control power bus. Loss of the normal power supply to this bus will cause an automatic transfer to the Unit 3 125-V distribution panel but it will also cause a loss of the dc power supply to instrumentation that will require Unit 2 to be shut down (ie, 125-V Distribution Panel I will be lost).

The proposed changes in the Technical Specifications will have a negligible effect on what is done at the plant. The requirements would reinforce conditions that already exist. The ac 480-V switch gears 28 and 29 are not connected during power operation, other than by accident (i.e., the failure to restore the ac system to its proper orientation after the situation described above). The DG 2/3 instrumentation and control power bus is not supposed to be on the Unit 3 125-V distribution system. In a PRA the conditions modeled are those that exist at the plant. Even though no Technical Specifications specifically prohibit the electric power configurations under question in this issue, the PRA would treat the human errors required for the systems to be in this configuration. The existence of the Technical Specifications recommended would not significantly affect the probabilities associated with these human errors.

6. Conclusion

Since the proposed Technical Specifications will not alter plant conditions, as modeled in a PRA, the effect on the risk due to core melt is negligible. The PRA model for Dresden Unit 2 would not change due to proposed Technical Specifications. The electric power configurations at issue here would still be attainable through human error and the probabilities associated with those errors would not change to any significant degree.

VI-10.A Testing of Reactor Trip System and Engineered Safety Features, Including Response Time

1. NRC Evaluation

Dresden-2 meets all current criteria for this topic, except for response time testing of the reactor trip system and the engineered safety features.

2. NRC Recommendation

A change should be made in the Dresden Technical Specifications to require response time testing of both the reactor trip system and the engineered safety features system. A test program should be implemented to fulfill the obligations required by the new Technical Specifications.

3. Systems Affected

The systems affected are the reactor protection system and all systems comprising the emergency safety features.

4. Comments

The important aspect of response time testing to be considered when examining its impact on risk is the relatively short time period involved in the response time test. The time involved in response time testing is normally of the order of a few seconds. When considering the functionability of a system in a risk analysis, the time constraints are considerably longer. The system may have a 1/2 hour or longer to perform its function. With this amount of time available, the short time periods measured in the response time test have no effect on system operability. The functional tests that are performed on the reactor protection system and the emergency safety features provide sufficient information to determine the operability of each system.

The operability tests will determine whether the systems are functioning or not. Whether the system functions fast enough to pass a response time test or not is not important, from a risk viewpoint, only that the system did respond. For example, a system may be required to automatically initiate within 10 seconds to pass a response time test. During the test the system starts within 15 seconds and fails the response time test. However, from a "risk due to core melt" viewpoint, the system has not failed because it did automatically initiate. This information could be obtained from the system operability tests.

The longer time periods for system actuation are based on the work done in PRAs for various plants including Millstone-1, Brown's Ferry, and others. In particular, the results for Dresden-2 can be expected to be very similar to those for Millstone-1.

5. Analysis

As stated in the comments section, response time testing deals with time periods on the order of seconds. The time periods that the systems have to respond to an initiating event to prevent a core melt, are on the order of several minutes. Response time testing does not provide significant additional information beyond normal system operability tests.

6. Conclusions

The requirement for response time testing does not have an effect on the risk due to a core melt.

VI-10.B Electrical, Instrumentation, and Control Portions of Shared Systems

1. NRC Evaluation

Dresden Unit 2 does not meet present NRC licensing requirements with regard to some of the shared EI&C features. The areas of concern include:

1. Single failures in conjunction with an accident at Unit 2 and a loss of offsite power can prohibit required safeguard systems from receiving power.

2. There are no interlocks or LCOs preventing the parallel operation of the shared 125-V and 250-V dc battery systems.

3. There are no interlocks or LCO requirements preventing the normal/bypass switches for the DG 2/3 from being in the "bypass" position during normal operation. (This is a concern with a single failure and a loss of offsite power.)

4. The 125-V and 250-V dc systems are shared.

5. Stored energy for diesel generator operation does not meet the 7-day minimum, or time to replenish, whichever is longer, required by current licensing criteria.

2. NRC Recommendation

The following four recommendations have been made concerning the issues noted above:

1. The Dresden-2 dc systems should be modified to prevent parallel operation of dc batteries.

2. A Technical Specification, preventing unit operation while the normal/bypass switch is in the bypass position, should be implemented.

3. The dc systems, although shared, do meet the single failure criterion and are therefore acceptable.

4. Additional onsite fuel oil storage should be provided.

3. Comments

The third item of the NRC Recommendation will not be evaluated here since the design has been deemed acceptable using the single failure criterion. Some issues have not been included in this analysis for one of several reasons. Either the item has been resolved and is no longer under consideration (this refers to Item 7 of Section V of the SER), the item is handled under another SEP topic, or the analysis of a particular item is beyond the scope of this analysis.

The information used in this analysis came primarily from discussions with plant personnel on operating procedures at the Dresden plant.

The normal/bypass switch on the "swing" diesel, diesel generator 2/3, is supposed to be in the normal position at all times during power operation of Dresden Unit 2. The only time the normal/bypass switch is supposed to be in the "bypass" position is during a loss of offsite power to both Dresden Unit 2 and Dresden Unit 3 where the dedicated diesel, diesel generator 2, does not start. Only under these conditions is the normal/bypass switch to be in the off-normal position.

When a PRA is performed, actual plant conditions are modeled as opposed to the conditions as set forth in the Technical Specifications. In this case the condition at the plant during normal operation is that the normal/bypass switch is in the "bypass" switch only when there has been an error of restoration. The existence of a Technical Specification regarding the position of this switch would not change the probability that the switch would be left in the wrong position. Since this event is modeled in the analysis and the Technical Specification does not change the model, this proposed change would not have any effect on the risk due to core melt as evaluated in a PRA.

The normal/bypass switch on the swing diesel is supposed to be in the normal position during power operation. The position of this switch is checked using two separate procedures. One procedure, DGP 1-S1 Unit 2 (3) Master Start Up Check List, is used every time the plant starts up. The second procedure is performed monthly (DOP 6600-4, Diesel Generator 2/3: Preparation for Standby Operation). With these checks already existing at Dresden-2, the Technical Specification changes suggested would not affect the availability of the diesel generator.

The NRC has recommended that additional fuel storage capability be provided onsite. According to the SEP Technical Evaluation Report, sufficient fuel for 2 days' operation of the diesels is provided onsite. In previous PRAs, WASH-1400, the Millstone-1 IREP study, etc., the accident analysis covered a period of 24 hours. The analyses were limited to this time frame since the risk due to a core melt is greatest during this time. Since almost all of the risk of a core melt occurs within 1 day, the 2-day supply of fuel oil at Dresden is large enough so that any additional fuel oil storage capacity would not have an effect on the plant risk. (In particular for the electric power system the probability of restoring offsite power within 2 days, making the diesels not essential, is very high. This, coupled with the reduced risks after, makes the need for extended operation of the diesels, beyond 2 days, highly unlikely.)

4. Systems Affected

The electric power systems (ac and dc) are the systems affected by this issue.

5. Analysis

As with the normal/bypass switch, the dc battery systems are not supposed to be operated in parallel during normal power operation. The two dc divisions would be paralleled only as a result of a restoration failure following maintenance on one of the dc batteries. Figure 1 is a fault tree that represents those failures of the dc power system that can occur only when the two divisions of the dc power system are connected. (This fault tree can be used to model the failures due to parallel operation of either the 125-V or 250-V dc power systems.) The system failures modeled include uncleared faults that can short both batteries and the overloading of one battery if all other sources of dc power, the battery chargers, and the other battery, fail. Loss of the ac power supply to the dc batteries is represented by the dominant mechanisms for loss of ac power to the dc buses. For normal power available this is a loss of the battery charger and for loss of normal power situations the diesel generator failures are included.

Data used in this analysis is derived from three sources. Battery and battery charger failure data is taken from NUREG-0666 "A Probabilistic Safety Analysis of DC Power Supply Requirements for

Nuclear Power Plants." The failure to restore the dc power system to normal is derived from human error probabilities as defined in NUREG/CR-1278 "Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications." The remaining data is from the Millstone IREP study. The data is shown in Table 1.

The failure to restore the dc system to normal is a combination of the battery failure rate and the human error associated with a failure to restore the system after maintenance. This human error is actually a double error since the restoration procedure is double-checked by an operator prior to returning the system to service.

The data used for the batteries is dependent on the resolution of Issues VIII-3.A and VIII-3.B. It was assumed that these issues would be resolved such that the testing requirements were met and that the dc power system was properly annunciated. The change in the system reliability, expressed as a percentage change, should be relatively independent of the resolution of these issues, however, since the additional failure probability due to parallel operation of the dc systems is being compared to the failure rate of the batteries.

The probability of failing the dc power system due to faults related to the potential paralleling of the two dc divisions is :.3E-9.

Procedures do exist that require the operator to transfer dc buses from one battery to the other to locate ground faults . If done properly, this procedure calls for the bus to be transferred back to its normal power supply in the same step in which the original transfer occurred. The coupling of these two actions in one step of the procedure makes the two events extremely dependent upon each other, i.e., if the bus is transferred, it is extremely likely that the operator will perform the second act and realign the system properly. The combination of events required for the use of these procedures to lead to a battery (dc system) failure are: 1) the existence of a ground fault, 2) a misuse of the procedures, and 3) a failure to discover this misuse.

In the Millstone IREP study, the misalignment of the dc buses was considered and assigned a probability of 1×10^{-3}. This number is at least as large as what would be expected at Dresden-2 due to the ground detection procedures (i.e., DOP 6900-6, 125 volt Ground Detection - Unit 2). The human errors associated with failure to follow a procedure correctly and for this error to go undetected are of this order of magnitude. Therefore, this misalignment of the dc buses can be expected to have no greater affect on the dominant sequences at Dresden than bus misalignment did on the Millstone IREP study. The bus misalignment did not appear in any dominant sequences in the Millstone IREP study. The possible misalignment of the dc buses due to a misuse of these procedures should not contribute to the dominant core melt sequences of Dresden-2.

The ground fault detection procedure used at Dresden-2 on the 250V system, is used on the average of once per month. During this procedure, the loads in one DC train are transferred to the other battery/battery charger combination. It is possible that during the performance of this procedure, voltage differences between the two DC power supplies could cause excessive current surges that would lead to interruption of the DC power supply to components supplied by the 250V system. The high currents could cause protective circuite beakers to open or fuses to blow.

To date, no such problem has occurred at Dresden-2 during the performance of the 250V ground fault detection procedure. Dresden-2 has been operational for 12 years, so the procedure has been utilized approximately 140 times. A zero failure approximation can be developed for the probability that something will happen during an upcoming use of this procedure. For a 50 percent confidence level, the following approximation is used:

$$\theta = \frac{\theta_2 \chi^2_{22}}{2(n+1)} \quad \text{where } \theta_2 = \frac{n+1}{T} \quad \text{and}$$

n = no. of occurrences of a failure during procedure use = 0

T = no. of times the procedure is used ≈ 140

χ^2 @ 50 percent confidence level = 1.386.

$$\Theta = \frac{140^{\frac{1}{1.386}}}{2(1)} = 5E\text{-}3/\text{procedure use.}$$

Since the procedure is used 12 times a year, this yields a frequency of approximately 6E-2/Ryr. for the possibility of a fault developing due to the proper use of this procedure.

As a transient initiator, this is not a significant figure. The expected transient rate at a nuclear power plant is approximately 7 per year. Also, this procedure is not likely to be used following the occurrence of an accident precursor so that a contribution to the existing accident sequences due to a fault created by this procedure does not exist.

However, there is one aspect of this event that is beyond the scope of this limited analysis. The possible effects of the fault created in the use of this procedure include: tripping the plant, degradation of the HPCI system and failure of the isolation condenser system. Different circuit breakers/fuses would have to fail to affect these events. If the worst case is assumed and all three events were to occur, the use of this procedure could lead to a significant accident sequence. Insufficient information is available to determine the livelihood of this worst case or of the different permutations of faults that may be caused by the use of this procedure to determine their contribution to the plant risk.

6. Conclusion

The resolution of this issue eliminates a negligible contributor to the risk due to core melt. The dominant sequences contain failure combinations that have a probability on the order of 10^{-5} or 10^{-6}. This combination of events, as analyzed here, is three orders of magnitude lower than the dominant contributors. Eliminating these contributions would have no effect on the risk due to core melt.

TABLE 1

Data Summary

Fault	Failure Rate	Fault Exposure Time	Unavailability
Battery fails	4.4E-3/yr	9 months[1]	3.3E-3
Improper restoration of circuit breakers after maintenance	5E-4/d		
a. Restoration improperly done	1E-2/d		
b. Error not detected during check	5E-2/d		
DC system not restored to normal following battery Maintenance			1.5E-7
Circuit breaker fails to remain open	1E-6/hr	9 months	6.5E-3
(2 in series)			4.2E-5
Uncleared fault on battery bus		---	3E-5
Battery charger fault			4E-6
DG fails to start/run			6E-2
Probability that transient was loss of offsite power		(.2/7) =	3E-2

[1]Assumes minimum test and annunciation requirements are met.

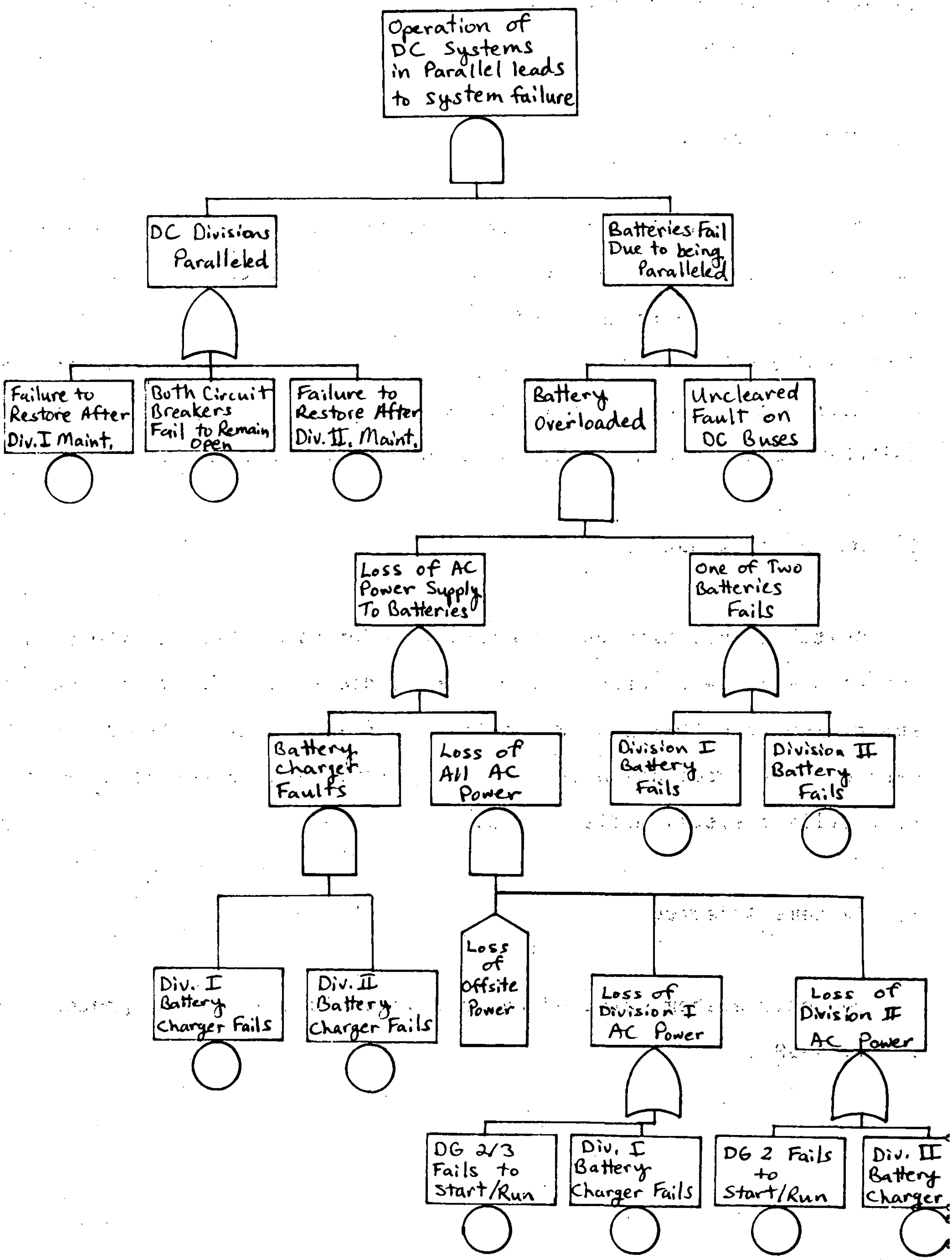

Figure 1. DC Power Parallel Failure Fault Tree

VII-1.A Isolation of Reactor Protection System From Nonsafety Systems

1. NRC Evaluation

Dresden meets all current criteria except that: 1) there are no isolation devices between the nuclear flux monitoring systems and the process recorders and indicating instruments, 2) isolation devices are not provided to isolate the APRM system from the process computer, and 3) the power supplies for the RPS channels are not suitably isolated and are not qualified as IE equipment.

2. NRC Recommendation

Suitable isolation devices should be provided between the flux monitors and the process recorder and indicating instrumentation, and between the APRM system and the process computer. The third part of this issue has been resolved through a licensee commitment to provide a system modification.

3. Systems Affected

The only system affected by this issue is the reactor protection system (RPS).

4. Comments

The flux monitoring system and APRM system are the only parts of the RPS that are affected by this issue. The level and pressure sensor inputs are all unaffected.

5. Analysis

The Dresden Unit 2 RPS is very similar to the Millstone-1 RPS as analyzed in the Millstone Unit 1 Interim Reliability Evaluation Program (IREP) report. The effects of the loss of the neutron flux monitoring system and the APRM system on the Dresden RPS would be identical to the effect on the Millstone-1 RPS.

The effect of inadequate isolation of these two signals from other plant equipment can be modeled, very conservatively, by assuming that a fault exists in the plant equipment that will fail both the flux monitoring and the APRM systems. By assigning a failure probability of 1.0 to the flux monitors and the APRM and evaluating the RPS fault tree, the effect of the inadequate isolation is modeled. An analysis using this data is very conservative since it is assumed that a fault exists in the process computer, process recorders, or indicating instrumentation, and further that the fault affects all channels of the flux monitoring and APRM systems. This is an extremely conservative assumption, but if it can be shown that using this assumption makes little or no difference in the RPS reliability then the resolution of the issue will have no effect on the RPS reliability.

The failure of the neutron flux monitoring and APRM systems will have no effect on the remaining instrumentation in the RPS. The level, pressure and other sensors will continue to operate independently of the two failed systems.

When the RPS fault tree is analyzed under these conditions, modeling Dresden as it is now, the RPS failure rate is calculated to be

$$P(RPS) = 8.9 \times 10^{-6}/d.$$

This failure rate must be compared to the RPS failure rate assuming that the neutron flux monitoring and APRM systems are adequately isolated. The normal, random, failure rates for the components in the systems are used in the RPS fault tree. Reevaluating the fault tree yields a failure rate of

$$P(RPS) = 8.9 \times 10^{-6}/d.$$

The two cases modeled have the same failure rate for the RPS. This is due to the total domination of common mode mechanical faults of the control rods in the failure of the RPS. The redundancy in the signal processing equipment is such that failure of even two independent signals has no effect on the system reliability.

6. Conclusion

Due to the relatively high (when compared to signal failures) probability of common mode mechanical faults involving the control rods the failure of the flux monitoring and APRM systems almost no effect on the failure rate of the RPS and consequently the risk due to core melt. Therefore the resolution of this issue will have no effect on the risk due to core melt at Dresden Unit 2.

VII-3. Systems Required for Safe Shutdown

1. NRC Evaluation

Procedures do exist to place the reactor into hot shutdown, hot standby and cold shutdown. However, these procedures do not rely entirely on safety grade systems. Additionally, no procedures exist for the operators to bring Dresden-2 to a cold shutdown condition from outside the control room.

2. NRC Recommendation

Procedures should be developed for shutdown and cooldown using only safety grade systems (HPCI, LPCI, CS, ESW, pressure relief valves, ac and dc power). Procedures for cooldown to cold shutdown from outside the control room should be developed.

3. Systems Affected

This issue affects the use of these systems:

High-Pressure Coolant Injection System
Low-Pressure Coolant Injection System
Isolation Condenser System
Core Spray System
Shutdown Cooling System
Pressure Relief System

4. Comments

The issue of not having procedures for reaching cold shutdown from outside the control room addresses the broad issue of how important it is to go from hot shutdown to cold shutdown as far as core melt risk is concerned. PRA studies to date have assumed that the dominant part of the risk is involved in getting to hot shutdown. These studies, in examining the accident scenarios where there is a failure to reach hot shutdown, have examined the major contributors to the risk due to core melt.

In examining the procedures for achieving hot shutdown a PRA will take into consideration all possible means by which the operators can be expected to reach hot shutdown. This would include methods using only safety grade systems and systems that are not safety grade provided their use can be justified. Justification usually is automatic system operation or a written procedure that details the use of the system. The potential lower reliability of a nonsafety grade system as opposed to a safety grade system would be evaluated on a system-by-system basis.

5. Analysis

The combination of low risk due to not achieving cold shutdown coupled with the probability of needing to reach this condition from outside the control room makes this issue insignificant as a contributor to risk due to core melt. Previous PRA studies, WASH-1400 and the Millstone-1 IREP study included, have not treated the ability to reach cold shutdown as a significant contributor to risk due to core melt. It is reasonable to expect this relationship to apply to Dresden Unit 2.

Although the ECCs are not mentioned in the procedures for normal shutdown, their absence does not necessarily have an impact on the risk analyzed in a PRA. These systems are referenced in the emergency procedures (Loss of Feedwater, Loss of AC Power, etc.) and credit is taken for their use in a PRA. The normal shutdown procedures do not recommend the use of safety grade systems only. However, the sequence of events that can lead to core melt certainly does not include normal shutdown. In those cases where the normal shutdown procedures cannot be followed it is reasonable to expect the operators to follow one of the emergency procedures. Through the combination of normal and emergency procedures the operators have the option of using any of the systems he can use to reach hot shutdown. Additionally, by restricting the shutdown procedures to safety grade systems several possible paths for reaching hot shutdown are lost. Although the nonsafety grade systems are not as reliable as the safety grade systems they do add some redundant capabilities. Recovering the nonsafety grade systems from the shutdown procedures would eliminate that redundancy.

6. Conclusion

The ability to use both safety grade and nonsafety grade systems in shutting down the Dresden Unit 2 plant provides redundant shutdown capabilities beyond those of the safety grade systems alone. The use of a procedure using only safety grade systems would not yield a reduction in the risk due to core melt.

The need to get from hot shutdown to cold shutdown is generally not addressed as significant to risk. Therefore, the existence or nonexistence of procedures to achieve cold shutdown from outside the control room should have little effect on the risk due to core melt.

VIII-2. Onsite Emergency Power Systems--Diesel Generator

1. NRC Evaluation

The standby diesel generator systems should be designed so that spurious operation of protective trips do not prevent the diesel generators from performing their function. This can be accomplished in two ways. All trips, except engine overspeed and generator differential, can be bypassed during operation or the trip must be implemented by two, or more, independent measurements for each trip parameter with coincident logic for trip actuation. The Dresden-2 under-frequency protective trip is not bypassed nor does it require coincident trip signals.

2. NRC Recommendation

The under-frequency protective trip should be bypassed during system operation.

3. Systems Affected

The system affected by this issue is the ac power system.

4. Comments

This issue deals with the fraction of diesel generator failures that can be attributed to spurious operation of the protective trip logic. The analysis will compare the reduced failure rate with the trips bypassed compared to the "overall" (including spurious trips) failure rate of the diesel generator.

5. Analysis

The diesel generators at Dresden-2 are 2500 kW General Motors generators. Failure data for these diesels are included in NUREG/CR-1362 Data Summaries of Licensee Event Reports of Diesel Generators at U.S. Commercial Nuclear Power Plants. In this data there are 94 recorded incidences of diesel generator failures for diesel generators of this size and make for the 3-year period of 1976-1978. Included in this data are 21 diesel generator failures at Dresden-2. Of these 94 LERs on diesel generators, eight were diesel trips and three were delayed starts. None of the failures were due to spurious under-frequency trips.

A zero failure approximation can be developed for the failure rate of the diesel generator due to spurious under-frequency trip operation using the above data. The 50 percent confidence level would be defined by

$$\theta = \frac{\theta_2 \chi^2_{22}}{2(N+1)}$$

where $\theta_2 = \frac{N+1}{T}$ and

N = no. of failures due to spurious under-frequency trips = 0

T = no. of diesel generator failures

(94-8-3)* = 83

χ^2 @ 50 percent confidence level = 1.386

This yields a failure rate of 8.3E-3 per diesel generator failure, ie, the under-frequency trips would contribute only 0.8 percent to the diesel generator failure rate.

The analysis of the Dresden-2 electric power system is based on the Millstone-1 Interim Reliability Evaluation Program (IREP) analysis. Although some modifications were made in the electric power fault tree, the failure rates for the diesel generators were generic data and can be applied directly to Dresden-2. The demand failure rate used for diesel generators was 6×10^{-2}/d. By eliminating the under-frequency contribution to this demand failure rate the diesel generator demand failure rate becomes 5.95×10^{-2}/d.

If this new number is used in the analysis of the modified Millstone electric power fault trees (representing Dresden-2) the

*The delayed start and diesel generator trips are not considered as diesel generator failures (the trips are not spurious trips).

effect is not noticeable in the dominant sequences. This is true even though failure of the electric power system is a dominant contributor in most of the dominant core melt sequences. Additionally, the difference between these two numbers, 6E-2 and 5.95E-2, is insignificant compared to the uncertainty involved in the diesel generator failure data itself.

6. Conclusion

Although the failure of the diesel generators and the electric power system is a dominant contributor to the risk due to a core melt, the change in the diesel generator failure rate produced by the resolution of this issue has no noticeable effect on the risk due to core melt. The relatively small, 0.8 percent, contribution of the under-frequency trip to the diesel generator failure rate does not make a noticeable contribution to the failure rate of the electric power system.

VIII-3.A. Station Battery Capacity Test Requirements

1. NRC Evaluation

The Dresden-2 station battery capacity tests do not conform to current licensing requirements. No periodic battery service tests are required by the Technical Specifications and the load discharge test does not appear to verify the required 80 percent capacity.

2. NRC Evaluation

Two types of tests should be performed on the Dresden-2 batteries. At least every 18 months a battery service test should be performed to verify that the battery capacity is adequate for two hours of emergency operation. At least every 60 months a battery discharge test should be performed to verify that the battery capacity is at least 80 percent of the manufacturer's rating. Both tests should be conducted during shutdown.

3. System Affected

The dc power system is the only system affected.

4. Comments

It is beyond the scope of this analysis to determine whether or not the battery tests performed at Dresden Unit 2 are equivalent to the requirements recommended by the NRC. If the tests are deemed to be equivalent then obviously there is no conflict and the issue is resolved. If the testing done is determined to be inadequate, ie, the tests do not provide sufficient information concerning the condition of the batteries, then the following analysis will prove useful. In regard to the development of data in a PRA, inadequate testing is treated as if no testing is performed. In this case, this is a conservative assumption.

5. Analysis

The effect of this issue is on the failure probability for the station batteries. An approximation for the failure probability for a routinely tested component, the battery, is

$$P(\text{batt}) \simeq \frac{1}{2} \lambda_o t$$

where:

$P(\text{batt})$ = probability of battery failure
λ_o = battery failure rate
t = time between tests

If the component, battery, is never tested the approximation becomes

$$P(\text{batt}) \simeq \lambda_o t$$

where t is the time the battery has been in service.

To meet the present licensing criteria, the test interval can be no more than 18 months. The battery failure data is taken from licensee event reports as compiled on NUREG-0666 "A Probabilistic

Safety Analysis of DC Power Supply Requirements for Nuclear Power Plants." In this report the battery failure rate is given as 8.7 x 10^{-3}/yr.

Using this data the probability of battery failure, battery tested every 18 months, is

$$\frac{1}{2}(1.5 \text{ yr})\ (8.7 \times 10^{-3}/\text{yr})$$
$$= 6.5\text{E-}3.$$

The Dresden Unit 2 has been in commercial operation for approximately 12 years. If there has been no adequate testing of the batteries the probability of battery failure becomes

$$(12 \text{ yrs})\ (8.7 \times 10^{-3}/\text{yr})$$
$$= 1.0\text{E-}1.$$

The battery reliability is improved by approximately a factor of 15. This factor of improvement is unchanged by the resolution of SEP Topic VIII-3.B. The issues raised by that topic will change the magnitude of the two failure probabilities (increased monitoring reduces the battery failure probability); however, the ratio between the two numbers remains unchanged.

The loss of dc power does have an impact on the dominant sequences that lead to core melt using either the model developed for the Millstone IREP study or that model as modified to represent Dresden Unit 2. As modeled the dc power system affected only a small portion of the dominant sequences. However, this model used data assuming adequate testing of the dc batteries. The increase in the battery failure rate due to a lack of testing would not only increase the probability of the dominant accident sequences in which dc power failures appear, but it would also affect sequences in which it does not at present appear.

6. Conclusions

The increase in the battery unavailability due to a lack of testing is greater than an order of magnitude (a factor of 15).* This increase would increase the probability of the dominant sequences in which battery failures occur having a significant impact on the risk due to core melt. As an example, one dominant sequence in the Millstone IREP study contributed ~3 percent of the total core melt probability. DC power faults contributed to only half of the dominant cut sets in this sequence. This sequence probability would be increased by a factor of 7.5, would contribute ~21 percent of the total core melt probability and thus would increase the core melt probability by almost 20 percent.

*Recall that we assume that there has been no effective battery testing for the operating life of the plant.

VIII-3.B. DC Power System Bus Voltage Monitoring and Annunciation

1. NRC Evaluation

The dc battery and bus monitoring system should provide adequate information to the operator so that he can determine battery and bus states and take corrective action if necessary. The Dresden Unit 2 system is deficient since it does not have control room indication of battery voltage, battery current, battery charger current, or fuse/breaker status.

2. NRC Recommendation

Instrumentation should be installed to provide control room indications and alarms for the status of the following dc power system parameters:

Battery voltage
Battery current (ammeter charge/discharge)
Battery charger output current
Battery breaker or fuse open alarm
Battery charger breaker or fuse open alarm

3. Systems Affected

The system affected by this issue is the dc power system.

4. Comments

This analysis is based on NUREG-0666 "A Probabilistic Safety Analysis of DC Power Supply Requirements for Nuclear Power Plants" and the Millstone-1 IREP study. The evaluation of detectable and undetectable battery faults is based on NUREG-0666. (Undetectable faults are those that are not detected until battery tests.) The importance of the dc power system relative to core melt risk is based on the Millstone-1 IREP study and a comparison of Millstone-1 and Dresden Unit 2.

The adequacy of the station battery tests is being evaluated as part of SEP Topic VIII-3.A. It can be assumed that with the resolution of Topic VIII-3.A the Dresden tests will provide the equivalent protection of 18-month battery service tests. Therefore, the effects of the difference in bus annunciations addressed in this issue will be evaluated assuming the current test criteria is met at Dresden-2.

5. Analysis

Figure 1 is a fault tree for loss of power at the main dc buses at Dresden Unit 2. The two buses are identical and the fault tree in Figure 1 can be used for either bus. The standby battery charger is not modeled in this fault tree since it is not normally connected to either bus.

This issue affects the probability that battery faults will go undetected between tests and the probability that given a demand on the battery, the breakers between the battery and the bus will be closed, allowing the battery to supply power to the bus. Any fault that goes undetected between tests will have a fault exposure time of one half the test interval. Faults detected immediately have a fault exposure time on the order of the duration of the challenge to the dc power system.

The data used to analyze the effects of the additional dc monitors is derived from several sources. These include NUREG-0666, the Millstone IREP study, and current battery test requirements. From NUREG-0666, approximately one half of the battery faults reported in Licensee Event Reports were discovered at battery tests even though the minimum monitoring requirements were met. An improved Dresden Unit 2 monitoring can not be expected to yield better results. Further, the assumption is made that no battery faults are detected until testing with the present monitoring system. The data used for this analysis is shown in Table 1.

From this data and the fault tree, the failure probability for each dc bus is 3.4E-4 without the proposed monitoring changes and 6.6E-5 with the proposed changes. The probability of a bus failure is reduced approximately a factor of 5.

6. Conclusions

In the Millstone IREP study the failure of the dc power system did affect some of the dominant sequences that lead to core melt, although the contribution was not very large. When the Millstone-1 fault trees were modified to reflect the differences between Millstone 1 and Dresden Unit 2 the effects of the failure of the dc power system remain as contributors to at least some of the dominant sequences. The additional monitoring proposed in this issue virtually eliminates the contribution of dc bus failures to the risk due to core melt.

It is of interest to note that for shorter test intervals the factor by which the bus failure probability is reduced is relatively unchanged. For test intervals of 3 months the factor is 4 (instead of 5). Particularly for yearly tests the bus failure probability is reduced from 2.1E-4 to 4.4E-5.

TABLE 1

DC Bus Failure Data

Fault	Failure Rate	Fault Exposure Time	Unavailability
Bus out of service for maintenance			
DC bus failure		1 hr	2×10^{-6}
Battery charger failure			4×10^{-6}
AC power failure			2×10^{-2}
Breaker fails to remain closed			
"as is"	1×10^{-6}/hr	9 months	6.5×10^{-3}
"modified annunciation"	1×10^{-6}/hr	1 hr	1×10^{-6}
Battery faults detectable			
"as is"	0	---	0.0
"modified monitoring"	4.4×10^{-3}/yr	1 hr	5×10^{-7}
Battery faults not detectable			
"as is"	8.7×10^{-3}/yr	9 months	6.5E-3
"modified monitoring"	4.4×10^{-3}/yr	9 months	3.3E-3
Breaker open for test/maint. on battery			

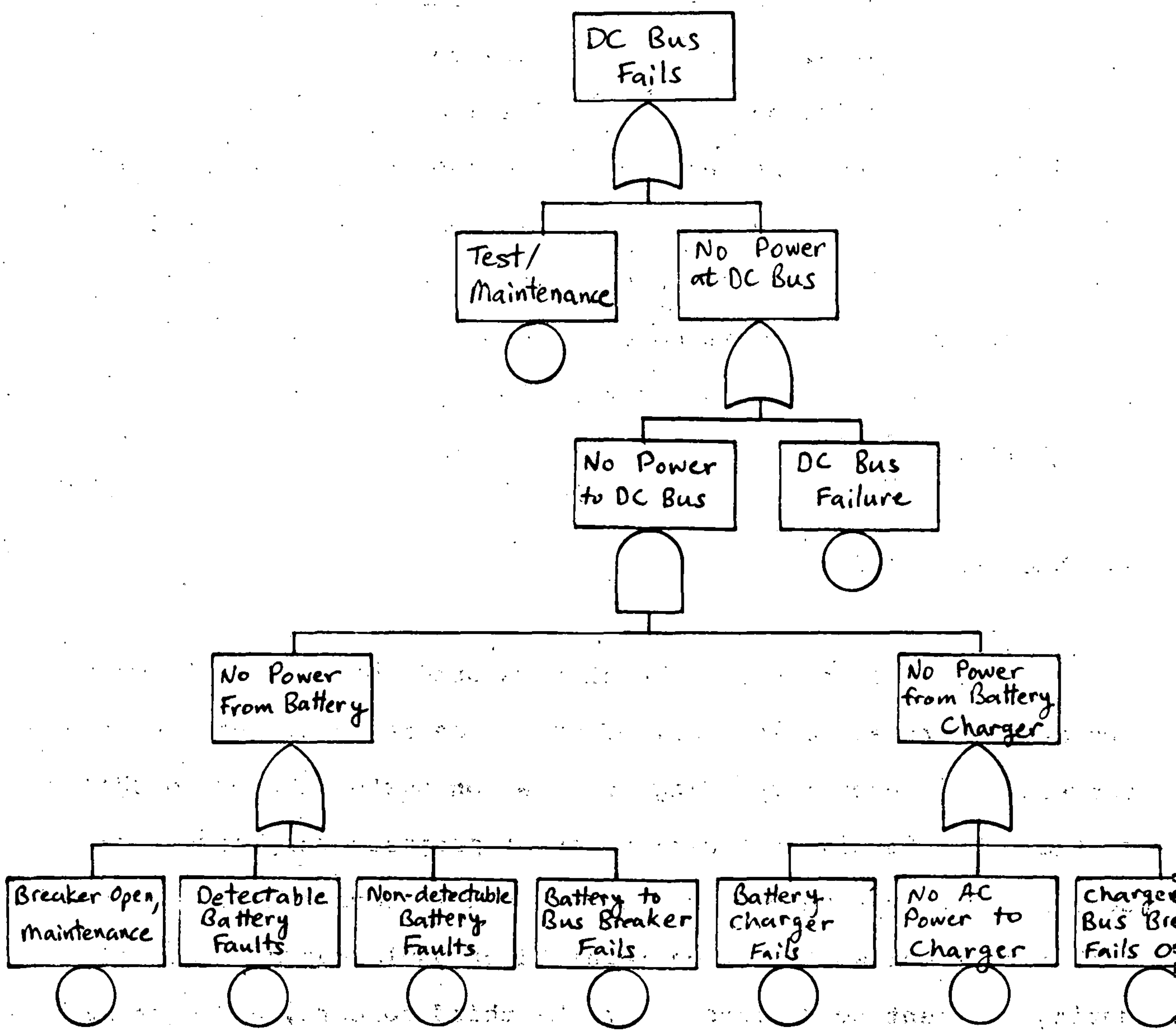

Figure 1. DC Bus Failure Fault Tree

IX-5 Ventilation Systems

1. NRC Evaluation

There are three subsets of concerns dealing with the ventilation systems at Dresden-2. First, there are several concerns about the adequacy of ventilation (cooling) of emergency equipment during accidents. Second, the ventilation system in the reactor building is not designed to always direct flow from areas of low radioactivity to areas of high radioactivity, as required by current criteria. Third, there is concern about hydrogen buildup in the battery room during loss of ventilation, possibly causing a detonation.

2. NRC Recommendation

For the first concern, the plant should determine if system operations are actually impaired by the deviations. For the second concern, the reactor building ventilation system should be changed so that flow is always toward areas of progressively higher radioactivity, or it should be demonstrated that access to the reactor building either would not be required or would not be impaired during accident conditions. For the third concern, hydrogen buildup in the battery room should be prevented.

3. Systems Affected

The cooling concerns affect several emergency systems. The radioactivity issue affects the ability of the operator to perform recovery actions requiring access to the reactor building.

4. Comments

The question of system failures due to loss of room cooling was addressed during the IREP Millstone-1 study. Our analysis here draws upon that experience.

The concern about battery room ventilation is that if the batteries are charged while the battery room ventilation is inoperative, a detonation may occur due to hydrogen buildup in the battery cells and/or room. Assessing the likelihood of this scenario and the consequences of a detonation are beyond the scope of this study.

5. Analysis

For the first concern, the adequacy of room cooling for emergency systems, we reference the study "Emergency Core Cooling System Corner Room Heatup Test*" performed by General Electric for Millstone-1. The results of this study indicate that emergency equipment at Millstone-1 could operate for at least 36 hours with no

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*Letter from R. W. Straub to Millstone-1, MS-2320, 2 May 1970.

room cooling without endangering any safety function. This question was further explored while performing the IREP Millstone-1 PRA and no potential system failures due to (support system) ventilation failures were identified. Based on our review of the configuration of the Dresden-2 plant, we have no reason to believe that these conclusions are not valid for Dresden-2, also.

For the second concern, access to the reactor building during accident conditions, we assess that, based on the IREP Millstone-1 PRA and the Dresden-2 fault trees, there is one recovery action for dominant accidents requiring such access. This is local operation of the Isolation Condenser makeup valve. Failure of Isolation Condenser makeup, without this recovery action, would be a dominant contributor to risk. However, if the operator could not enter the Reactor Building because of high radiation levels to open this valve, then core damage would have already occurred, creating the radiation source, and recovery action would not be effective. That is, if the operator does not open this valve before core damage and thus before any radiation is spread in the reactor building, then it is too late to prevent core melt.

6. Conclusions

The Dresden-2 emergency systems should not be susceptible to failure due to loss of ventilation. The operator should not be prevented from entering the reactor building until after core damage has already occurred and then it is too late for recovery actions. Thus this issue is of low importance to risk.

XV-1. Decrease in Feedwater Temperature, Increase in Feedwater Flow, and Increase in Steam Flow

1. NRC Evaluation

Failure of the feedwater controller to maximum demand results in an increase in reactor power and vessel inventory. A feedwater control failure at rated power is similar to the turbine trip event at rated power with the turbine bypass operable. However, for the feedwater controller event, the turbine trip signal occurs when the reactor is at above rated power. Hence, this event can be limiting with respect to minimum critical power and is evaluated in reload analysis. To meet current criteria, surveillance of the turbine bypass system is required. Since the bypass system was assumed to operate in the analysis of this event, limitations to either reactor power or minimum critical power ratio would be required in the Technical Specifications to cover the case where the bypass system is found inoperable.

2. NRC Recommendations

Perform surveillance of the turbine bypass system and write limitations to either reactor power or the minumum critical power ratio in the Technical Specifications to cover the case where the bypass system is found inoperable.

3. System Affected

This event is a transient initiating event for core melt sequences.

4. Comments

This event had been considered when performing the analysis of initiating events in the Millstone-1 IREP PRA. That analysis applies directly here.

5. Analysis

The risk significance of <u>any</u> transient with the turbine bypass unavailable is that this makes the power conversion system (PCS) unavailable as a system to be used for heat removal during the transient. The transient initiators in the Millstone-1 IREP Study were grouped according to whether the PCS was available since that was the only mitigating system found to be affected by transients. The transients studied were those identified in the document EPRI NP-801 and a failure modes and effects analysis (FMEA) of postulated transient initiators in support systems.

The specific case of a transient with the turbine bypass unavailable was treated as a transient with <u>subsequent</u> loss of the power conversion system. One reason for loss of the power conversion system is turbine bypass failure, so including these transients with transients causing loss of the PCS would result in double-counting these transients.

The Millstone-1 transient frequency with loss of the PCS (initially or subsequently) is 2.14/yr, dominated by MSIV closure, loss of condenser vacuum, increasing feedwater flow, and pressure regulator failing open. Note that increasing feedwater flow causes loss of the PCS independently of turbine bypass failure. However, as stated above, the risk significance of this issue extends beyond this one transient. The key point is that transients involving turbine bypass failure do not contribute to loss of the PCS.

This analysis shows that the historical rate of turbine bypass unavailability has been small enough compared to other causes of loss of the PCS that even if the proposed limitations on reactor operation with the turbine bypass unavailable prevented transients under that condition, the effect on the overall transient rate with loss of the PCS would be negligible.

6. Conclusion

Requiring limitations on reactor operation with the turbine bypass unavailable would have no effect on risk because loss of the turbine bypass does not significantly contribute to the unavailability of the power conversion system compared to other causes.

XV-16 Radiological Consequences of Small Lines Carrying Primary Coolant Outside Containment

XV-18 Radiological Consequences of a Main Steam Line Failure Outside Containment

1. NRC Evaluation

These two issues address exceeding 10 CFR Part 100 doses during events which do not lead to core melt.

2. NRC Recommendations

Make whatever changes necessary to prevent exceeding 10 CFR Part 100 doses.

3. Systems Affected

These issues affect offsite consequences.

4. Comments

None

5. Analysis

PRAs have shown that the overwhelmingly dominant portion of the risk from nuclear power plants is from core melt accidents. The contribution from rod ejection, spent fuel pool accidents, transportation accidents, and other small dose releases is negligible compared to the massive releases of radioactive material from core melt accidents. Thus the effect on risk of resolving these issues is negligible.

6. Conclusion

These two issues have no effect on risk.

V. Dresden-2 Fault Trees

Following are the modifications made to the Millstone-1 IREP PRA fault tree to represent the risk model for Dresden-2. These changes include logic changes to the fault trees and data changes. Table V-1 gives the Millstone-1 fault trees, which are modified, in the order in which the modifications are given below. In addition, Dresden-2 has a High Pressure Injection System instead of a Feedwater Coolant Injection System, which Millstone-1 has. For the purposes of qualitative examination, the analysis and fault tree for the High Pressure Injection System (which is similar to the Dresden-2 High Pressure Injection System) used in the IREP Brown's Ferry PRA have been reproduced here.

APPENDIX E

REFERENCES TO CORRESPONDENCE FOR EACH TOPIC EVALUATED

SEP Topic No.	Date	Reference
II-1.A	10/6/81	Letter from D. M. Crutchfield (NRC) to L. DelGeorge (CECo), Subject: SEP Topics II.1.A, Exclusion Area Authority and Control, and II-1.B, Population Distribution (Dresden 2).
II-1.B	10/6/81	See reference for Topic II-1.A.
II-1.C	8/20/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic II-1.C, Potential Hazards Due to Nearby Transportation, Institutional, Industrial and Military Facilities - Dresden Unit 2.
II-2.A	4/24/81	Letter from D. M. Crutchfield (NRC) to J. S. Abel (CECo), Subject: SEP Topic II-2.A, Severe Weather Phenomena.
II-2.C	8/26/81	Letter from D. M. Crutchfield (NRC) to L. DelGeorge (CECo), Subject: SEP Topic II-2.C, Atmospheric Transport and Diffusion Characteristics for Accident Analysis - Dresden Unit 2.
II-3.A	9/16/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Hydrology Topics II-3.A, II-3.B, II-3.B.1 and II-3.C, Dresden Nuclear Power Station, Unit No. 2.
II-3.B	9/16/82	See reference for Topic II-3.A.
II-3.B.1	9/16/82	See reference for Topic II-3.A.
II-3.C	9/16/82	See reference for Topic II-3.A.
II-4	7/9/81	Letter from D. M. Crutchfield (NRC) to J. S. Abel (CECo), Subject: SEP Review Topics II-4, Geology and Seismology, and II-4.B, Proximity of Capable Tectonic Structures in Plant Vicinity.
II-4.A	6/8/81	Letter from D. M. Crutchfield (NRC) to all SEP Owners, Subject: Site Specific Ground Response Spectra for SEP Plants Located in the Eastern United States.
II-4.B	7/9/81	See reference for Topic II-4.
II-4.C	6/8/81	See reference for Topic II-4.A.
II-4.D	6/30/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Safety Topic II-4.D, Stability of Slopes - Dresden Nuclear Power Station, Unit No. 2.

SEP Topic No.	Date	Reference
II.4.E	9/3/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic II-4.E, Dam Integrity - Dresden Unit 2.
II-4.F	6/30/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic II-4.F, Settlement of Foundations and Buried Equipment - Dresden Nuclear Power Station, Unit No. 2.
III-1	9/2/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo) Subject: SEP Topic III-1, Quality Group Classification of Components and Systems - Dresden Nuclear Power Station Unit 2.
	7/7/81	Letter from D. M. Crutchfield (NRC) to J. S. Abel (CECo), Subject: SEP Topic VII-3, Safe Shutdown Systems Safety Evaluation Report (Dresden Power Station, Unit No. 2).
III-2	9/21/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic III-2, Wind and Tornado Loadings - Dresden Nuclear Generating Station, Unit No. 2.
III-3.A	6/4/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic III-3.A, Effects of High Water Level on Structures - Dresden Nuclear Power Station Unit 2.
III-3.C	6/30/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: Dresden 2 Nuclear Power Station, Unit No. 2, Safety Evaluation Report on SEP Topic III-3.C, Inservice Inspection of Water Control Structures.
III-4.A	6/28/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic III-4.A, Tornado Missiles - Dresden 2.
III-4.B	9/16/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic III-4.B, Turbine Missiles - Dresden Nuclear Power Station, Unit 2.
III-4.C	6/28/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic III-4.C, Internally Generated Missiles - Dresden 2.
III-4.D	2/22/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic III-4.D, Site Proximity Missiles (Including Aircraft) - Dresden 2.

SEP Topic No.	Date	Reference
III-5.A	9/21/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: Dresden Nuclear Power Station, Unit No. 2 - SEP Topic III-5.A, Effects of Pipe Break on Structures, Systems and Components Inside Containment.
III-5.B	8/20/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic III-5.B, Pipe Break Outside Containment - Dresden Nuclear Power Station Unit 2.
III-6	6/30/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Safety Topics III-6, Seismic Design Consideration, and III-11, Component Integrity - Dresden Nuclear Power Station Unit No. 2.
III-7.B	9/21/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic III-7.B, Design Codes, Design Criteria and Load Combinations - Dresden Nuclear Power Station, Unit No. 2.
III-7.D	4/13/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic III-7.D, Containment Structural Integrity Test - Dresden Nuclear Power Station Unit 2.
III-8.A	2/26/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: Systematic Evaluation Program Topic III-8.A, Loose Parts Monitoring and Core Barrel Vibration Program - Dresden 2.
III-8.C	6/29/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: Dresden Nuclear Power Station, Unit 2 - SEP Topic III-8.C, Irradiation Damage, Use of Sensitized Stainless Steel and Fatigue Resistance.
III-10.A	6/26/81	Letter from D. M. Crutchfield (NRC) to J. S. Abel (CECo), Subject: SEP Topic III-10.A, Thermal Overload Protection for Motors of Motor-Operated Valves, Safety Evaluation Report for Dresden Unit 2.
III-10.C	5/22/79	Letter from D. L. Ziemann (NRC) to C. Reed (CECo), Subject: Topic III-10.C, Dresden Nuclear Power Station, Unit No. 2.
IV-1.A	10/26/81	Letter from D. M. Crutchfield (NRC) to L. DelGeorge (CECo), Subject: SEP Topic IV-1.A., Operation With Less Than All Loops in Service - Dresden 2.
IV-2	12/4/81	Letter from T. J. Rausch (CECo) to D. M. Crutchfield (NRC), Subject: Dresden 2 SEP Topics: IV-2, Reactivity Control Systems NRC Docket No. 50-237.

SEP Topic No.	Date	Reference
IV-3	1/22/82	Letter from D. M. Crutchfield (NRC) to L. DelGeorge (CECo), Subject: SEP Topic IV-3, BWR Jet Pump Operating Indications - Dresden 2.
V-4	6/29/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic V-4, Piping and Safe-End Integrity - Dresden Nuclear Power Station Unit 2.
V-5	6/23/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic V-5, Reactor Coolant Pressure Boundary Leakage Detection, Dresden Nuclear Power Station Unit No. 2.
V-6	9/3/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic V-6, Reactor Vessel Integrity - Dresden Unit 2.
V-10.A	11/8/79	Letter from D. L. Ziemann (NRC) to D. L. Peoples (CECo), Subject: Completion of SEP Topic V-10.A - Dresden 2.
V-10.B	4/24/81	Letter from D. M. Crutchfield (NRC) to J. S. Abel (CECo), Subject: Dresden 2 - SEP Topics V-10.B, RHR Reliability; V-11.B, RHR Interlock Requirements; and VII-3, Systems Required for Safe Shutdown (Safe Shutdown Systems Report).
V-11.A	7/10/81	Letter from D. M. Crutchfield (NRC) to J. S. Abel (CECo), Subject: SEP Topics V-11.A, Requirements for Isolation of High and Low Pressure Systems and V-II.B, RHR Interlock Requirements - Safety Evaluation Report for Dresden Unit 2.
V-11.B	4/24/81	See reference for Topic V-10.B.
	7/10/81	See reference for Topic V-11.A.
V-12.A	4/16/81	Letter from D. M. Crutchfield (NRC) to J. S. Abel (CECo), Subject: SEP Topic V-12.A, Water Purity of Boiling Water Reactor Primary Coolant - Dresden Nuclear Power Station, Unit Nos. 1 and 2.
VI-1	6/30/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic VI-1, Organic Materials and Post Accident Chemistry, Dresden Nuclear Power Station, Unit No. 2.

SEP Topic No.	Date	Reference
VI-2.D	8/19/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: Systematic Evaluation Program (SEP) for Dresden Nuclear Power Station, Unit 2 - Evaluation Report on Topics VI-2.D and VI-3 (Docket No. 50-237).
VI-3	8/19/82	See reference for Topic VI-2.D.
VI-4	9/24/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic VI-4, Containment Isolation System - Dresden Nuclear Power Station, Unit 2.
	6/23/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic VI-4, Containment Isolation System (Electrical), Final Safety Evaluation Report For Dresden Unit 2.
VI-6	6/25/82	Letter from D. Eisenhut (NRC) to L. DelGeorge (CECo), Subject: Dresden Nuclear Power Station, Unit 2 and 3, regarding exemptions to Section 50.54 and Appendix J to 10 CFR Part 50.
VI-7.A.3	11/3/81	Letter from T. J. Rausch (CECo) to D. M. Crutchfield (NRC), Subject: Dresden 2 SEP Topics: VI-7.A.3, ECCS Actuation Systems, and VII-2, Engineered Safety Features System Control Logic and Design.
VI-7.A.4	4/23/82	Letter from D. M. Crutchfield (NRC) to L. DelGeorge (CECo), Subject: SEP Topic VI-7.A.4 - Core Spray Nozzle Effectiveness - Dresden 2.
VI-7.C	3/13/81	Letter from D. M. Crutchfield (NRC) to J. S. Abel (CECo), Subject: SEP Topics VI-7.C, ECCS Single Failure Criterion and Requirements for Locking Out Power to Valves, and VI-7.C.2, Failure Mode Analysis (Dresden Unit 2).
VI-7.C.1	2/5/82	Letter from D. M. Crutchfield (NRC) to L. DelGeorge (CECo), Subject: SEP Topic VI-7.C.1, Appendix K - Electrical, Instrumentation and Control (EIC) Re-Reviews (Dresden Nuclear Power Station Unit No. 2).
VI-7.C.2	3/13/81	See reference for Topic VI-7.C.
VI-7.D	10/18/78	Letter from M. S. Turbak (CECo) to D. K. Davis (NRC), Subject: Dresden Station Units 1 and 2 Review of Eight SEP Topics - NRC Docket Nos. 50-10/237.

SEP Topic No.	Date	Reference
VI-10.A	9/2/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic VI-10.A, Testing of Reactor Trip System and Engineered Safety Features, Including Response Time Testing, Revised Safety Analysis Report For Dresden Unit 2 Nuclear Power Station.
VI-10.B	9/24/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic VI-10.B, Electrical, Instrumentation and Control Portions of Shared Systems - Safety Evaluation Report for Dresden Nuclear Power Station, Unit 2.
	11/3/81	Letter from T. J. Rausch (CECo) to D. M. Crutchfield (NRC), Subject: Dresden 2 SEP Topics: VI-10.B, Shared Systems for Multiple Unit Stations.
VII-1.A	8/5/82	Letter from D. M. Crutchfield (NRC) to L. DelGeorge (CECo), Subject: SEP Topic VII-1.A, Isolation of Reactor Protection System From Non-Safety Systems, Including Qualification of Isolation Devices, Draft Safety Evaluation For Dresden Unit 2.
VII-1.B	10/18/78	See reference for Topic VI-7.D.
VII-2	11/3/81	See reference for Topic VI-7.A.3.
VII-3	4/24/81	See reference for Topic V-10.B.
	7/7/81	Letter from D. M. Crutchfield (NRC) to J. S. Abel (CECo), Subject: SEP Topic VII-3, Safe Shutdown Systems Safety Evaluation Report (Dresden Power Station, Unit No. 2).
VII-6	8/26/81	Letter from D. M. Crutchfield (NRC) to L. DelGeorge (CECo), Subject: SEP Topic VII-6, Frequency Decay Safety Evaluation for Dresden 2.
VIII-1.A	6/15/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic VIII-1.A, Potential Equipment Failures Associated With a Degraded Grid Voltage - Dresden Unit 2.
VIII-2	6/26/81	Letter from D. M. Crutchfield (NRC) to J. S. Abel (CECo), Subject: SEP Topic VIII-2, Onsite Emergency Power Systems, Diesel Generators, Safety Evaluation for Dresden 2.

SEP Topic No.	Date	Reference
VIII-3.A	7/7/81	Letter from D. M. Crutchfield (NRC) to J. S. Abel (CECo), Subject: SEP Topic VIII-3.A, Station Battery Capacity Test Requirements, Safety Evaluation for Dresden 2.
VIII-3.B	6/26/81	Letter from D. M. Crutchfield (NRC) to J. S. Abel (CECo), Subject: SEP Topic VIII-3.B, DC Power System Bus Voltage Monitoring and Annunciation, Safety Evaluation for Dresden Unit 2.
VIII-4	11/30/81	Letter from D. M. Crutchfield (NRC) to L. DelGeorge (CECo), Subject: SEP Topic VIII-4, Electrical Penetrations of Reactor Containment - Safety Evaluation Report (Dresden Unit 2).
IX-1	5/5/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic IX-1, Fuel Storage - Dresden 2.
IX-3	11/3/81	Letter from T. J. Rausch (CECo), to D. M. Crutchfield (NRC), Subject: Dresden 2 SEP Topics: IX-3, Station Service and Cooling Water System.
IX-5	9/2/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: Forwarding Evaluation Report of SEP Topic IX-5, Ventilation Systems for the Dresden Nuclear Power Plant Unit 2.
IX-6	6/23/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic IX-6, Fire Protection.
XIII-2	4/14/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic XIII-2, Safeguards/Industrial Security.
XV-1	12/29/81	Letter from D. M. Crutchfield (NRC) to L. DelGeorge (CECo), Subject: Dresden 2 - SEP Topic XV-1, Decrease in Feedwater Temperature, Increase in Feedwater Flow and Increase in Steam Flow.
XV-3	12/7/81	Letter from D. M. Crutchfield (NRC) to L. DelGeorge (CECo), Subject: SEP Topics XV-3, XV-4, and XV-19 (Systems).
XV-4	12/7/81	See reference for Topic XV-3.
XV-5	12/15/81	Letter from D. M. Crutchfield (NRC) to L. DelGeorge (CECo), Subject: Dresden 2 - SEP Topics XV-5, Loss of Normal Feedwater Flow, and XV-9, Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate.

SEP Topic No.	Date	Reference
XV-7	1/4/82	Letter from D. M. Crutchfield (NRC) to L. DelGeorge (CECo), Subject: Dresden 2 - SEP Topics XV-7 and XV-15.
XV-8	2/8/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: Dresden 2 - SEP Topic XV-8, Control Rod Misoperation; Topic XV-11, Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (BWR); and Topic XV-13, Spectrum of Rod Drop Accidents (Systems).
XV-9	12/15/81	See reference for Topic XV-5.
XV-11	2/8/82	See reference for Topic XV-8.
XV-13	2/8/82	See reference for Topic XV-8.
	12/28/81	Letter from D. M. Crutchfield (NRC) to L. DelGeorge (CECo), Subject: Dresden 2 - SEP Topic XV-13, Spectrum of Rod Drop Accidents.
XV-14	12/15/81	Letter from D. M. Crutchfield (NRC) to L. DelGeorge (CECo), Subject: Dresden 2 - SEP Topic XV-14, Inadvertent Operation of Emergency Core Cooling Systems (ECCS) That Increase Reactor Coolant Inventory.
XV-15	1/4/82	See reference for Topic XV-7.
XV-16	10/20/81	Letter from T. J. Rausch (CECo), to D. M. Crutchfield (NRC), Subject: Dresden 2 - SEP Topic XV-16, Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment, NRC Docket 50-237.
XV-18	9/2/82	Letter from P. O'Connor (NRC) to L. DelGeorge (CECo), Subject: SEP Topic XV-18, Radiological Consequences of a Main Steam Line Failure Outside Containment, Dresden Nuclear Power Station, Unit 2.
XV-19	12/7/81	See reference for Topic XV-3.
	1/5/82	Letter from D. M. Crutchfield (NRC) to L. DelGeorge (CECo), Subject: Dresden 2 - SEP Topic XV-19, Radiological Consequences of Loss-of-Coolant Accident From Piping Breaks Within the Reactor Coolant Pressure Boundary.

SEP Topic No.	Date	Reference
XV-20	11/20/81	Letter from D. M. Crutchfield (NRC) to L. DelGeorge (CECo), Subject: Dresden-2, SEP Topic XV-20, Radiological Consequences of Fuel Damaging Accident.
XVII	11/20/79	Letter from D. L. Ziemann (NRC) to D. L. Peoples (CECo), Subject: Completion of SEP Topic XVII - Operational QA Program, Dresden 2.

APPENDIX F

SECTIONS OF OAK RIDGE NATIONAL LABORATORY REPORT
PERTAINING TO DRESDEN UNIT 2

ORNL/NSIC-199

Nuclear Safety Information Center

Engineering Technology Division

REVIEW OF THE OPERATING EXPERIENCE HISTORY OF DRESDEN UNIT 2 THROUGH 1981 FOR THE NUCLEAR REGULATORY COMMISSIONS'S SYSTEMATIC EVALUATION PROGRAM

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EXECUTIVE SUMMARY

The Systematic Evaluation Program Branch of the Nuclear Regulatory Commission (NRC) is conducting the Systematic Evaluation Program (SEP) for the purpose of determining the safety margins of the design and operation of ten of the older operating commercial nuclear power plants in the United States. These ten plants are being reevaluated in terms of pressnt NRC licensing requirements and regulations. Thus, the SEP is intended:

1. to establish documentation that shows how these ten plants compared with current acceptance criteria and guidelines on significant safety issues and to provide a technical rationale for acceptable departures from these criteria and guidelines,
2. to provide the capability for making integrated and balanced decisions with respect to any required backfitting, and
3. to provide for the early identification and resolution of any potential safety deficiency.

The SEP evaluates specific safety topics based on an integrated review of the overall ability of a plant to respond to certain design-basis events including normal operation, transients, and postulated accidents.

As part of the SEP, the NRC contracted with the Oak Ridge National Laboratory to perform operating history reviews. These reviews are intended to augment the SEP's safety topic review and to aid in the determination of priorities for required backfitting during the integrated assessment. Each review includes collection and evaluation of availability and capacity factors, forced shutdowns, forced power reductions, reportable events, environmental events, and radiological release events.

This summary presents the results from the review of the operating experience of the Dresden Unit 2 Nuclear Power Plant, which is a General-Electric-designed boiling-water reactor, owned and operated by Commonwealth Edison Company. The plant is located at Morris, Illinois. The reactor has a licensed thermal power of 2537 MW(t) and a design electric rating of 794 MW(e). Dresden 2 achieved initial criticality on January 7, 1970, and began commercial operation on June 9, 1972.

From 1970 through 1981, the cumulative reactor availability factor at Dresden Unit 2 was 77.1% and the cumulative unit capacity factor was 57.0%, while the average values for these two factors were 73.5 and 53.8%, respectively. The reactor availability was above average and the unit capacity factor was average. From 1973 through 1980, the reactor availability factor and unit capacity factor averaged 80.3 and 61.4%, respectively. The values were lower during 1970 and 1971 due to the introduction of spurious signals into the scram circuitry and maintenance outages to perform repairs on the main transformer and on the main turbine. In 1981, the refueling outage that began at the start of the year was extended to effect further repairs on the main turbine.

The operating history review focused on data evaluation which was divided into two segments: (1) evaluation of forced shutdowns and power reductions, and (2) evaluation of reportable events. Design basis events

(DBEs), which are defined in the NRC's *Standard Review Plan*,[1] are failures that initiate system transients and challenge engineered safety features. In the forced shutdown and power reduction segment, the review identified DBEs and recurring events that might indicate a potential operating concern. In the reportable event segment which included environmental events and radiological release events, the review identified significant events and recurring events that might indicate a potential operating concern. Significant events were either DBEs or events with a loss of engineered safety function.

Forced Shutdowns and Power Reductions

Of the 206 forced shutdowns and power reductions between 1970 and 1981, 68 were identified as DBEs of one of the following 11 types:

1. turbine trip (20),
2. loss of normal feedwater (10),
3. inadvertent closure of main steam isolation valve (MSIV) (9),
4. feedwater system malfunctions resulting in increased feedwater flow (8),
5. loss of condenser vacuum (7),
6. loss of external electric load (5),
7. single and multiple recirculation pump trips (3),
8. inadvertent opening of turbine bypass valves resulting in increased steam flow (2),
9. inadvertent opening of safety relief valve (2),
10. startup of an idle recirculation pump (1), and
11. control rod maloperation (1).

There are two aspects of the DBEs relative to frequency of occurrence:

1. The number of feedwater system malfunctions resulting in increased feedwater flow is somewhat higher than the experience of other plants with seven of the eight events occurring between 1970 and 1973. The causes for the eight events included the introduction of spurious signals by various types of personnel into the feedwater system circuitry (3), flow or level spikes caused by operators (3), feedwater control valve stuck open (1), and a blown fuse in a feedwater valve control circuit.
2. The total number of DBEs taken individually for each event type with the exception of the feedwater malfunctions is consistent with the experience of other plants; however, of the 11 DBEs experienced at Dresden Unit 2, 5 different DBE types occurred 7 or more times which overall represents an increased number for a variety of DBE types as compared with other plants.

Of the 68 DBEs identified through 1981, 40 occurred between 1970 and 1973 at a time when Dresden Unit 2 was experiencing problems with the introduction of spurious signals (either by personnel or inherently) in various parts of the reactor protection system. In all instances except one, the engineered safety features functioned properly to bring the reactor to a safe shutdown.

The one event where engineered safety features failed to function properly occurred on June 5, 1970, and involved a series of multiple failures complicated by operator error and procedural inadequacies. With the reactor undergoing initial startup tests and operating at 75% power (623 MWe), the incident was initiated by a spurious signal generated in the electrohydraulic control of the turbine-generator set which caused the turbine control valve to open further and the steam bypass valves to the condenser to fully open. Within one second the turbine tripped and the reactor scrammed. The two operating feedwater pumps tripped due to low suction pressure caused by the increased feedwater flow. Subsequently, the MSIVs closed and the water level control in the pressure vessel became difficult. Water level began rising again, but because the level-indicator chart pen being observed by the operator stuck, the operator further increased the flow rate of feedwater not knowing the level was still increasing. By the time the operator discovered the stuck pen, the water level had risen enough to flood the main steam lines and the isolation condenser steam line. The incident was further complicated at this point by a lack of procedural guidance under conditions of high reactor coolant in the pressure vessel. The continued input of water coupled with afterheat from the reactor core and closure of the main-steam-line valves caused the pressure-vessel pressure to begin increasing rapidly. The isolation condenser system was actuated manually, but it was shut off automatically due to a too-low trip setting of the condensate-return-line flow required by an erroneous technical specification. An attempt to reopen the main-steam-line valves to dump steam through the turbine-bypass valves failed because the valves had not been reset from the earlier trip that had closed them. Following the automatic tripping of the recirculation pumps and automatic startup of the standby diesel generators, the low pressure spray- and coolant-injection systems started but did not inject water because the reactor pressure exceeded the pump head of both systems. The high-pressure coolant-injection system started but did not inject water, because it had been valved out earlier for repairs after proof-testing its backup system (automatic depressurization system) as provided for in the technical specifications. Actual water injection by this system would have been automatically inhibited by the high-water signal from the pressure-vessel water-level monitors. With the isolation condenser inoperable, the operator manually opened a pressure-relief valve several times throughout the incident to dump steam to the pressure-suppression pool to reduce the pressure in the pressure vessel in order to remove the reactor decay heat. The high-pressure coolant released from this valve impinged on the lifting levers of two other safety valves and partially opened them. They remained open until they were closed manually after the vessel was depressurized and cooled down. Several thousand gallons of primary water leaked to the drywell. The containment zone was contaminated, but no measurable radioactivity was released to the site or the environs. Damage to the plant was minor.

Reportable Events

In the reportable event segment of the operating history review of Dresden 2, 625 events were reviewed. The trend for the number of reportable events submitted by Dresden 2 has remained relatively constant since 1974 with an average of 62 events from 1974 through 1981. For this 8-year period, a low of 45 events were reported in 1980 with the peak years being 1977 and 1981 with 71 and 74 events reported, respectively. For 1970 through 1973, an average of 33 events per year were reported. Inherent equipment failures caused 51% of the reportable events. Human error (including administrative, design, fabrication, installation, maintenance, and operator error) either caused or were directly related to 49% of the reportable events. Of the reports involving human error, the three dominant categories were maintenance errors (11.6%), administrative control errors (11.1%), and design errors (9.4%). There is no apparent trend in the causes of the reportable events.

Of the 631 reportable events, 19 are considered significant:

- pipe cracks in various systems (6),
- loss of emergency power – both diesels (3),
- MSIVs fail to close (2),
- reactor depressurization and subsequent leakage of reactor coolant water to the torus (1),
- isolation condenser rendered inoperable (1),
- safety relief valve fails to close (1),
- forty-six control rods fail to insert completely (1),
- scram discharge volume high level alarm fails (1),
- failure of an automatic depressurization system valve in conjunction with the high pressure coolant injection (HPCI) system inoperable resulting in loss of high pressure safety function (1),
- reactor protection system signal fails to alarm on turbine control valve closure (1), and
- thirty loose restraining clamp bolt keepers found on 19 of 20 jet pumps.

The major contributor to the significant event types was an assortment of various equipment and component failures which caused 9 of the 19 significnt events. Pipe cracks that were identified from 1974 through 1978 in assorted systems accounted for six of the significant events. Several cracks were found in the coolant recirculation system and recirculation bypass loop while cracks were also found in the core spray system, feedwater system, and containment isolation systems. Pipe cracking is a generic BWR problem and the Pipe Cracking Study Group[2] formed by NRC has indicated stress-corrosion cracking as the cause. Three events were due to human error – two operator errors and one fabrication error.

Of the 19 significant events, 6 occurred in 1974, 5 of which involved the aforementioned pipe cracks. Other than 1974, no more than three significant events were noted in any one year with only one event categorized as significant for each of the years 1979, 1980, and 1981.

Recurring Events

The following nine types of recurring events were noted during the two segments of operating history review:

1. pipe cracks,
2. MSIV failures,
3. feedwater regulator valve problems,
4. diesel generator failures,
5. control rod and rod drive malfunctions,
6. radioactive water management/health physics program problems,
7. operator errors,
8. turbine control problems (valves and electro-hydraulic control system oil leaks), and
9. HPCI failures.

Three of the event types (pipe cracks, MSIV failures, and feedwater regulator valve problems) were identified either by Dresden Unit 2 or NRC and corrective measures undertaken. Three event types (diesel generator failures, control rod/rod drive probelms, and the radioactive waste management/health physics program problems) were identified by Dresden 2 and NRC and corrective actions taken or are under consideration, but these event types continue to recur and/or still are of concern. The remaining three event types continued to recur through 1981.

The problem concerning pipe cracks was discussed earlier in the reportable events section.

The MSIV failures involved inadvertent closures and failures to close. Various equipment and component modifications were made including modifications to the instrument air system to improve MSIV reliability and reduce inadvertent closures. The three failures to close occurred very early in the Dresden 2 operating experience and involved fouling of pilot valves due to particulates in the air supply to the pilot valves. Corrective action included blowing down the air supply line. The topic of MSIV failure to close was examined by NRC in late 1981.[3]

Feedwater regulator valve problems were limited to a 3-year period from 1973-1976. Problems involved seal leaks, valve stems breaking, and a blown fuse in a control circuit. Low reactor coolant water level resulted in five of the six events, but there were no instances of total loss of feedwater.

There were 44 instances where the diesel generators failed upon demand with 22 of these being failures to start. From 1975 to 1979, the diesel generators failed almost nine times per year compared to an average of two failures per year for other years. The increased frequency of failures could be partially attributed to an increased test frequency from monthly to weekly which began in December 1977. Estimates of failure rates on demand indicate Dresden 2 has experienced a failure rate greater than the median value found in the Reactor Safety Study[4] but within the upper bound of the study. An investigation into problems associated with the air starting system during 1979 resulted in a modification to the air starting circuitry which allows multiple start attempts prior to locking out the starting sequence. The previous design allowed only one start attempt before the starting sequence was locked out. Only one failure to start was reported in 1980 and none in 1981. However, in 1981 there were two diesel generator failures (October 23 and December 1) involving check valve failures on the Unit 2/3 diesel generator engine cooling systems (this is a single diesel generator that can be electrically aligned to serve either Unit 2 or Unit 3). A similar failure occurred on Unit 3 diesel on November 19, 1981. It was determined that the check valves on both diesel generator cooling water pumps had broken free of the pivot arm. These failures were not adequately characterized by operator observations and instrument readings during diesel generator surveillance tests, but were discovered by direct inspection of the internals of the check valve. It is not known how long these check valves were broken before their condition was detected since the broken valve discs were free to move within the valve bodies and may have been that way for some time before coming to rest in a position which would restrict flow enough to cause the diesel to trip on high engine temperature.[5] Dresden Unit 2 has experienced three instances of loss of emergency power from the diesel generator failures where in one of the three events one of the two inoperable diesels was restarted immediately and was declared operable.

Control rod and rod drive problems were prominent in the years 1970 (5), 1974 (5), 1977 (7), 1980 (8), and 1981 (6). Slow control rod insert time was experienced during the plant's early life; however, a design modification to the control rod drive inner filter corrected this. Uncoupling of one or two control rods dominated the failures for 1974, 1977, and 1980. The failures in 1981 involved excessive insert times.

The radioactive waste management/health physics program problems fall into two caterogies: (1) activity limits were exceeded in various radwaste and drain tanks and (2) a variety of equipment failures resulted in both gaseous and liquid leaks which, along with breakdowns in operations that involved health physics considerations, provided either the potential for exposures to personnel or resulted in exposures to personnel. The 17 instances where activity levels were exceeded involved sample tanks, radwaste tanks, floor drain tanks, above ground tanks, etc. The limit for activity level in these tanks is 0.7 Ci and the levels measured ranged from 0.74 to 5.6 Ci. The releases associated with the associated leaks were in most cases confined to within the plant itself. Those releases to plant environs were inconsequential. The most significant occurrences in the second category were two instances where control rods were withdrawn during control rod tests with the reactor shut down for refueling where personnel were within the line of sight of the core and two incidents involved overexposures. The first and more significant radiation overexposure occurred March 5, 1981, during maintenance work when a portion of a large radiation shielding plug was being removed from inside the reactor vessel during maintenance work. There was no fuel in the reactor. Because of an inaccurate instrument, plant personnel believed that the water level in the reactor was higher than it actually was; the lower water level did not provide adequate shielding for the highly radioactive reactor components beneath the shielding plug. The individual involved, a contractor employee, received an exposure of 21 rems while guiding a crane in the removal of the shielding plug. The second overexposure involved a contractor employee who received a cumulative radiation exposure of 3.02 rems for the time period January 1 to March 20, 1981. Although the total exposure was only slightly above the NRC limit of 3 rems per calendar quarter, NRC's Office of Inspection and Enforcement proposed fines for both incidents because this exposure, along with the larger exposure, indicated a serious weakness in the plant's program to control personnel radiation exposures.[7]

Operator errors, which include control room personnel, auxiliary operators, and maintenance and testing personnel (as compared to the broader category of human errors defined previously), directly caused or complicated some 36 forced reactor shutdowns and 20% of the reportable events. The errors were of three basic types: (1) during surveillance testing, personnel introduced spurious signals into various control and protection systems resulting in scrams — all but 3 of 11 of these type events occurred between 1970 and 1975, (2) during maintenance activities maintenance personnel were either troubleshooting particular problems or returning instrument/sensing lines to service and their actions resulted in tripping the reactor, and (3) incorrect or inadvertent personnel actions led directly to scramming the reactor, e.g., startup of an idle recirculation pump, inserting control rods too quickly, valving errors, transfer of control functions from automatic to manual, improper use of walkie-talkie resulting in actuation of turbine overspeed circuit signal causing a reactor trip, and jarring of an instrument rack. The operator error was the complicating cause in the June 5, 1970 depressurization event when the operator continued to focus his attention on the stuck level indicator for the water level in the pressure vessel and increased feedwater flow resulting in increased pressure in the pressure vessel.

Turbine control valve and turbine electro-hydraulic control (EHC) system problems occurred in clusters approximately every 3 to 4 years beginning in 1972, 1975 to 1976, and 1980. The control valve problems involved inherent failures and steam leaks. The EHC problems involved low oil pressure and oil leaks that led to forced shutdowns.

Fifty reportable events were filed that involved the HPCI. Fifteen of these represented failures of the HPCI on demand. The principal causes of failures were due to failures of motor-operated valves, the turbine stop valve, and the isolation valves. The HPCI failures were evenly distributed throughout Dresden Unit 2's operating experience although no failures on demand occurred in 1981. Estimates of the failure rate of the HPCI system for Dresden 2 indicate a failure rate several times that predicted in the Reactor Safety Study[4] and a factor of two greater than that observed from historical data.[6]

Conclusion

For this analysis of the operating history at Dresden 2, 206 forced shutdowns and power reductions were reviewed, along with 631 reportable events and other miscellaneous documentation concerning the operation of the Dresden Unit 2 Nuclear Power Plant. The objective was to indicate those areas of plant operation that compromised plant safety. This review identified one significant challenge to plant safety and six problems that should be of continued concern.

The most serious plant challenge to plant safety occurred on June 5, 1970 while Dresden Unit 2 was undergoing power testing and was operating at approximately 75% power, a spurious signal in the reactor pressure-control system altered the steam flow to the turbine and caused a turbine trip followed by a reactor scram. Subsequent erratic water-level and pressure control in the reactor vessel, compounded by a stuck indicator pen on a water-level monitor-recorder and inability of the isolation condenser to function as needed, led to discharge of steam and water through safety valves into the reactor dry well. No significant amount of radioactive contamination was discharged to the environment. There was no pressure damage of the reactor vessel or the dry-well containment walls.

The six areas of operation that should be of continued concern consist of two types: (1) those identified by either Dresden 2 or NRC and continue to recur — diesel generator failures, control rod and rod drive malfunctions, and radioactive waste management/health physics program problems, and (2) those areas of operation that have not been directly addressed by Dresden 2 or NRC — operator errors, turbine control valve and EHC problems, and HPCI failures. All six event types have continued to recur throughout Dresden 2's operating history.

References

1. U.S. Nuclear Regulatory Commission, "Accident Analysis for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapt. 15 of *Standard Review Plan*, NUREG-0800, July 1981.

2. U.S. Nuclear Regulatory Commission, *NRC Pipe Cracking Study Group, Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of BWR Plants*, NUREG-75/067, October 1975.

3. U.S. Nuclear Regulatory Commission, "Main Steam Isolation Valve Failures to Close," IE Circular 81-14, November 5, 1981.

4. U.S. Nuclear Regulatory Commission, *Reactor Safety Study*, WASH-1400 (NUREG 75/014), October 1975.

5. U.S. Nuclear Regulatory Commission, "Check Valve Failures on Diesel Generator Engine Cooling System," IE Information Notice 82-08, March 26, 1982.

6. *Precursors to Potential Severe Core Damage Accidents: 1969-1979 A Status Report*, NUREG/CR-2497 (ORNL/NSIC-182), June 1982.

7. "Dresden Fine Proposed," *Nuclear Safety*, Vol. 22, No. 6, pg. 803, November-December 1981.

ABSTRACT

A review of the operating experience of Dresden Unit 2 nuclear power plant through 1981 was performed by the staff of the Nuclear Safety Information Center for the Nuclear Regulatory Commission's Systematic Evaluation Program (SEP). Under the Commission's SEP Program the safety margins of the design and operation of the 11 oldest operating commercial nuclear power plants in the United States are being reevaluated.

The review of the operating experience for each plant included data collection and evaluation of availability and capacity factors, forced shutdowns, power reductions, reportable events (reportable occurrence, licensee event reports, etc.), and environmental considerations. As well, the review methodology and procedures as used in the review and evaluation are discussed. Data and information collected for forced shutdowns, power reductions, and reportable events are presented in appendices.

REVIEW OF THE OPERATING EXPERIENCE HISTORY OF DRESDEN 2 THROUGH 1981 FOR THE NUCLEAR REGULATORY COMMISSION'S SYSTEMATIC EVALUATION PROGRAM

1. SCOPE OF REVIEW

The assessment of the operating experience review for Dresden 2 covered the time from initial criticality through 1981. The data collection and evaluation included the following aspects of operation: availability and capacity factors, forced shutdowns and power reductions, reportable events, events of environmental importance and radioactivity releases, and evaluation of the operating experience in total. Tables at the end of Chap. 1 show the codes assigned to operational aspects of forced shutdowns, power reductions, and reportable events. These codes are used in the reporting of data collected during the review of operating experience.

1.1 Availability and Capacity Factors

Both reactor and unit availability factors were compiled for all years. Starting with 1974, the unit capacity factors using the design electrical rating (DER) in net megawatts (electric) and the maximum dependable capacity (MDC) in net megawatts (electric) were compiled as well. Data for the capacity factors were not available from earlier years.

The two availability and two capacity factors are defined as follows:

1. reactor availability =

$$\frac{\text{hours reactor critical + reactor reserve shutdown hours}}{\text{period hours}}$$

2. unit availability =

$$\frac{\text{hours generator on line + unit reserve shutdown hours}}{\text{period hours}}$$

3. $$\text{unit capacity (DER)} = \frac{\text{net electrical energy generated}}{\text{period hours x DER net}} \times 100 ,$$

4. $$\text{unit capacity (MDC)} = \frac{\text{net electrical energy generated}}{\text{period hours x MDC net}} \times 100 .$$

Reserve shutdown hours are the amounts of time the reactor is not critical or the unit is shutdown for administrative or other similar reasons when operation could have been continued.

1.2 Review of Forced Shutdowns and Power Reductions

Forced shutdowns and power reductions were reviewed, and data were collected on each incident. Scheduled shutdowns for refueling and maintenance were not included in the review. However, if a utility had a refueling outage scheduled, the plant experienced a shutdown as a result of an abnormal event prior to the scheduled refueling, the utility reported that the refueling was being rescheduled to coincide with the current shutdown, and the utility reported the cause of the shutdown as refueling, then this shutdown was considered as forced. Only that portion of the outage time concerned with the abnormal event, not the refueling time, was included in the compilations.

The power reductions were included to provide information and details that may have been associated with a previous or subsequent shutdown. The power reductions are included in the proper chronological sequence with the shutdowns in the data tables for the forced shutdowns and power reductions (see Appendixes).

The following data were compiled annually for the forced shutdowns and power reductions:

1. date of occurrence,
2. duration (hours),
3. power level (percent),
4. notation of whether the shutdowns were also reportable events [e.g., a licensee event report (LER) or abnormal occurrence report (AOR)],
5. summary description of events associated with the forced shutdown or power reduction,
6. cause of shutdown (Table 1.1),
7. method of shutdown (Table 1.1),
8. system taken from NUREG-0161 (Ref. 1) that was directly involved with the shutdown or power reduction (Table 1.2),
9. component directly involved with the shutdown or power reduction (Table 1.3), and
10. categorization of the shutdown or power reduction.

Each shutdown or power reduction was placed in one of two sets of significance categories. The shutdowns and power reductions were first evaluated against criteria for DBEs as described in Chap. 15 of the *Standard Review Plan*.[2] If the shutdown or power reduction could not be categorized as a design-basis initiating event, then it was placed in one of a series of Nuclear Safety Information Center (NSIC) categories. For further discussions of the two sets of significance categories, use of the categories, and a listing of them, see Sect. 3.1.

The listings for the cause, shutdown method, system involved, and component involved along with their respective codes are those used in the NUREG-0020 series[3] ("Gray Books") on shutdowns. Note that the information listed under the "System involved" column in the data tables in the appendixes indicates (1) a general classification of systems (fully written out) and (2) a specific system, which is coded with two letters, within the general classification.

1.3 Review of Reportable Events

The operating events as reported in LERs and LER predecessors [e.g., abnormal occurrence reports (AORs*), unusual event reports, reportable occurrences (ROs)] were reviewed. These types of reportable events were retrieved from the NSIC computer file. Approximately six years ago, operating experience information for operating nuclear power plants was input to the NSIC file for the period of time before LERs was reviewed. Any documents that contained LER-type information (such as equipment failures or abnormal events) were coded or indexed so that they could be retrieved in the same manner as an LER. Primarily, this involved various types of operating reports and general correspondence for the late 1960s and early 1970s.

The following information was recorded for each reportable event reviewed:

1. LER number or other means of identification of report type,
2. NSIC accession number (a unique identification number assigned to each document entered into the NSIC computer file),
3. date of the event,
4. date of the report or letter transmitting the event description,
5. status of the plant at the time of the occurrence (Table 1.4),
6. system involved with the reportable event (Table 1.2),
7. type of equipment involved with the reportable event (Table 1.5),
8. type of instrument involved with the reportable event (Table 1.5),
9. status of the component (equipment) at the time of the occurrence (Table 1.4),
10. abnormal condition associated with the reportable event (e.g., corrosion, vibration, leak) (Table 1.6),
11. cause of the reportable event (Table 1.4), and
12. significance of the reportable event.

As a step in the evaluation process, each reportable event was screened using the criteria further discussed in Sect. 3.2.

Note that in the tables of reportable events in Appendix A for Yankee Rowe, comments and/or details on the events were included.

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*The AO designation used by some utilities for identifying operational events during a particular time frame is <u>not</u> to be confused with those safety-significant events listed in the <u>Report to Congress on Abnormal Occurrences</u> (NUREG-0090 series) which also uses the AO designation.

1.4 Events of Environmental Importance and Releases of Radioactivity

Any significant or recurring environmental problems were summarized based on the review of forced shutdowns, power reductions, reportable events (environmental LERs), and operating reports. Routine radioactivity releases were tabulated as well, and releases where limits were exceeded were reviewed and are discussed in Sect. 4.5.1.4.

1.5 Evaluation of Operating Experience

The operating history of the plants was evaluated based on a review that involved screening, categorizing, and compiling data. Judgments and conclusions were made regarding safety problems, operations, trends (recurring problems), or potential safety concerns. Events were analyzed to determine their safety significance from the information provided through the various operating reports and the review process. The final safety analysis reports provided specific plant and equipment details when necessary.

Table 1.1. Codes and causes of forced shutdown or power reduction and methods of shutdown

Causes	
A	Equipment failure
B	Maintenance or testing
C	Refueling
D	Regulatory restriction
E	Operator training and license exams
F	Administrative
G	Operational error
H	Other
Methods	
1	Manual
2	Manual scram
3	Automatic scram
4	Continuation
5	Load reduction
9	Other

Table 1.2. Codes and systems involved with the forced shutdown, power reduction, or reportable event

System	Code
Reactor	RX
Reactor vessel internals	RA
Reactivity control systems	RB
Reactor core	RC
Reactor coolant and connected systems	CX
Reactor vessels and appurtenances	CA
Coolant recirculation systems and controls	CB
Main steam systems and controls	CC
Main steam isolation systems and controls	CD
Reactor core isolation cooling systems and controls	CE
Residual heat removal systems and controls	CF
Reactor coolant cleanup systems and controls	CG
Feedwater systems and controls	CH
Reactor coolant pressure boundary leakage detection systems	CI
Other coolant subsystems and their controls	CJ
Engineered safety features	SX
Reactor containment systems	SA
Containment heat removal systems and controls	SB
Containment air purification and cleanup systems and controls	SC
Containment isolation systems and controls	SD
Containment combustible control systems and controls	SE
Emergency core cooling systems and controls	SF
Core reflooding system	SF-A
Low-pressure safety injection system and controls	SF-B
High-pressure safety injection system and controls	SF-C
Core spray system and controls	SF-D
Control room habitability systems and controls	SG
Other engineered safety feature systems and their controls	SH
Containment purge system and controls	SH-A
Containment spray system and controls	SH-B
Auxiliary feedwater system and controls	SH-C
Standby gas treatment systems and controls	SH-D
Instrumentation and controls	IX
Reactor trip systems	IA
Engineered safety feature instrument systems	IB
Systems required for safe shutdown	IC
Safety-related display instrumentation	ID
Other instrument systems required for safety	IE
Other instrument systems not required for safety	IF
Electric power systems	EX
Offsite power systems and controls	EA
AC onsite power systems and controls	EB
DC onsite power systems and controls	EC
Onsite power systems and controls (composite ac and dc)	ED
Emergency generator systems and controls	EE
Emergency lighting systems and controls	EF
Other electric power systems and controls	EG

Table 1.2 (continued)

System	Code
Fuel storage and handling systems	FX
New fuel storage facilities	FA
Spent-fuel storage facilities	FB
Spent-fuel pool cooling and cleanup systems and controls	FC
Fuel handling systems	FD
Auxiliary water systems	WX
Station service water systems and controls	WA
Cooling systems for reactor auxiliaries and controls	WB
Demineralized water makeup systems and controls	WC
Potable and sanitary water systems and controls	WD
Ultimate heat sink facilities	WE
Condensate storage facilities	WF
Other auxiliary water systems and controls	WG
Auxiliary process systems	PX
Compressed air systems and controls	PA
Process sampling systems	PB
Chemical, volume control, and liquid poison systems and controls	PC
Failed-fuel detection systems	PD
Other auxiliary process systems and controls	PE
Other auxiliary systems	AX
Air conditioning, heating, cooling, and ventilation systems and controls	AA
Fire protection systems and controls	AB
Communication systems	AC
Other auxiliary systems and controls	AD
Steam and power conversion systems	HX
Turbine-generators and controls	HA
Main steam supply systems and controls (other than CC)	HB
Main condenser systems and controls	HC
Turbine gland sealing systems and controls	HD
Turbine bypass systems and controls	HE
Circulating water systems and controls	HF
Condensate cleanup systems and controls	HG
Condensate and feedwater systems and controls (other than CH)	HH
Steam generator blowdown systems and controls	HI
Other features of steam and power conversion systems (not included elsewhere)	HJ
Radioactive waste management systems	MX
Liquid radioactive waste management systems	MA
Gaseous radioactive waste management systems	MB
Process and effluent radiological monitoring systems	MC
Solid radioactive waste management systems	MD

Table 1.2 (continued)

System	Code
Radiation protection systems	BX
Area monitoring systems	BA
Airborne radioactivity monitoring systems	BB
Other	XX
Not applicable	ZZ

Table 1.3. Components involved with the forced shutdown or power reduction

Component type	Including
Accumulators	Scram accumulators Safety injection tanks Surge tanks
Air dryers	
Annunciator modules	Alarms Bells Buzzers Claxons Horns Gongs Sirens
Batteries and chargers	Chargers Dry cells Wet cells Storage cells
Blowers	Compressors Gas circulators Fans Ventilators
Circuit closers/interruptors	Circuit breakers Contactors Controllers Starters Switches (other than sensors) Switchgear
Control rods	Poison curtains
Control rod drive mechanisms	
Demineralizers	Ion exchangers
Electrical conductors	Bus Cable Wire
Engines, internal combustion	Butane engines Diesel engines Gasoline engines Natural gas engines Propane engines
Filters	Strainers Screens
Fuel elements	
Generators	Inverters
Heaters, electric	

Table 1.3 (continued)

Component type	Including
Heat exchangers	Condensers Coolers Evaporators Regenerative heat exchangers Steam generators Fan coil units
Instrumentation and controls	
Mechanical function units	Mechanical controllers Governors Gear boxes Varidrives Couplings
Motors	Electric motors Hydraulic motors Pneumatic (air) motors Servo motors
Penetrations, primary containment air locks	
Pipes, fittings	
Pumps	
Recombiners	
Relays	
Shock suppressors and supports	
Transformers	
Turbines	Steam turbines Gas turbines Hydro turbines
Valves	Valves Dampers
Valve operators	
Vessels, pressure	Containment vessels Dry wells Pressure suppression Pressurizers Reactor vessels

Table 1.4. Codes for data collected on plant status, component status, and cause of reportable events

Code	Plant status	Component status	Cause of reportable event
A	Construction	Maintenance and repair	Administrative error
B	Operation	Operation	Design error
C	Refueling	Testing	Fabrication error
D	Shutdown		Inherent error
E			Installation error
F			Lightning
G			Maintenance error
H			Operation error
I			Weather

Table 1.5. Codes for equipment and instruments involved in reportable events

Code		Code	
	Equipment		
A	Accumulator	W	Internal combustion engine
B	Air drier	X	Motor
C	Battery and charger	Y	Nozzle
D	Bearing	Z	Pipe and pipe fitting
E	Blower and dampers	AA	Power supply
F	Breaker	BB	Pressure vessel
G	Cables and connectors	CC	Pressurizer
H	Condenser	DD	Pump
I	Control rod	EE	Recombiner
J	Control rod drive	FF	Seal
K	Cooling tower	GG	Shock absorber
L	Crane	HH	Solenoid
M	Demineralizer	II	Steam generator
N	Diesel generator	JJ	Storage container
O	Fastener	KK	Support structure
P	Filter/screen	LL	Transformer
Q	Flange	MM	Tubing
R	Fuel element	NN	Turbine
S	Fuse	OO	Valve
T	Generator	PP	Valve, check
U	Heat exchanger	QQ	Valve operator
V	Heater		
	Instrumentation		
A	Alarm	L	Power range instrument
B	Amplifier	M	Pressure sensor
C	Electronic function unit	N	Radiation monitor
D	Failed fuel detection instrument	O	Recorder
E	Flow sensor	P	Relay
F	In-core instrument	Q	Seismic instrument
G	Indicator	R	Solid state device
H	Intermediate range instrument	S	Start-up range instrument
I	Level sensor	T	Switch
J	Meteorological instrument	U	Temperature sensor
K	Position instrument		

Table 1.6. Codes used for reportable events—abnormal conditions

	Mechanical
AA	Normal wear/aging/end of life: expected effect of normal usage
AB	Excessive wear/clearance: component (especially a moving component) experiences excessive wear or too much clearance or gap exists because of overuse, lack of lubrication
AC	Deterioration/damage: component is no longer at an acceptable level of quality (e.g., high temperature causes rubber seals to chemically break down or deteriorate, insulation breaks down)
AD	Break/shear: structural component physically breaks apart (not when something "breaks down")
AE	Warp/bend/deformation: shape of component is physically distorted
AF	Collapse: tank or compartment has an external pressure exerted that results in deformation
AG	Seize/bind/jam: component has inhibited movement caused by crud, foreign material, mechanical bonding, another component
AH	Excessive mechanical loads: mechanical load exceeds design limits
AI	Mechanical fatigue: failure due to repeated stress
AJ	Impact: the result of the force of one object striking another
AK	Improper lubrication: insufficient or incorrect lubrication
AL	Missing/loose: component is missing from its proper place or is loose or has undesired free movement
AM	Wrong part: incorrect component installed in a piece of equipment
AN	Wrong material: incorrect material used during fabrication or installation
AO	Weld-related failure: failure caused by defective weld or located in the heat-affected zone
AP	Vibration other than flow induced: vibration from any cause other than fluid flow
AQ	Crud buildup: buildup of foreign material such as dust, sticks, trash (not corrosion
AR	Corrosion/oxidation: unanticipated attack
AS	Dropped: component is dropped (includes control rod that is "dropped" into core)
AT	Leak, internal, within system: leak from one part of a system to another part of the same system
AU	Leak, internal, between systems: leak from one system to a different system
AV	Crack: defect in a component does not result in a leak through the wall

Table 1.6 (continued)

AW	Leak, external: defect in a component results in a leak from the system that is contained in an onsite building
AX	Leak to environment: leak not resulting from a cracked or broken component
AY	Was opened/transfers open: component is/was opened by error or spuriously opens
AZ	Was closed/transferred closed: component is/was wrongly closed by error or spuriously closes
BA	Fails to open: component is in the closed state <u>and</u> fails to open on demand (e.g., the circuit breaker "fails to open" when an overcurrent occurs)
BB	Fails to close: component is in the open state <u>and</u> fails to close on demand
BC	Malposition or maladjustment: component is out of desired position (e.g., normally open valve is closed) or adjusted improperly (not for instrument drift or out of calibration)
BD	Failure to start/turn on: component fails to start on demand
BE	Stopped/failed to continue to run: component fails to continue running when it has previously started
BF	Tripped: component <u>automatically</u> trips on or off (desired or undesired) (e.g., the turbine tripped because of overspeed, the circuit breaker tripped because of overspeed, or the circuit breaker tripped because of overload)
BG	Deenergized/power removed: component on system loses its driving potential but not necessarily electrical power [e.g.; (1) a fuse blows and there is no power to a sensor, and the sensor is deenergized; (2) a valve closes off the steam supply to a turbine, and the turbine has no driving power]
BH	Energized/power applied: component or system gains its driving potential but not necessarily electrical power (e.g., valve is opened allowing steam to turn a turbine)
BI	Unacceptable response time: component does not respond to a demand within a desired time frame but does not otherwise fail (e.g., a diesel generator fails to come to full speed within the time constraint)
BJ	High pressure: higher than normal or desired pressure exists in a component or system (<u>does</u> <u>not</u> include instrument misindications)

Table 1.6 (continued)

BK	Low pressure: lower than normal or desired pressure exists in a component or system (<u>does</u> <u>not</u> include instrument misindication)
BL	High temperature: component experiences a higher than normal or desired temperature
BM	Low temperature: component (or system) experiences a lower than normal or desired temperature
BN	Freezing: fluid medium (e.g., water) freezes in or on a component
BO	Excessive thermal cycling: frequent changes in temperature that could result in metal fatigue or cracking
BP	Unacceptable heatup/cooldown rate: heatup or cooldown rate exceeds limits
BQ	Thermal transient: system experiences an undesired or unstable thermal transient or thermal change
BR	Excessive number of pressure cycles: system experiences an undesired number of significant pressure changes (e.g., pressure pulses as from a positive displacement pump)
BS	High level/volume: higher than normal or desired level or volume exists (actual or potential) in a component, such as tank or sump, or area, such as auxiliary building (not for instrument misindication)
BT	Low level/volume: lower than normal or desired level or volume exists in a component (not for instrument misindication)
BU	Abnormal concentration/pH: an abnormal (either high or low) concentration of a chemical or reagent exists in a fluid system or an abnormal pH exists (does not include abnormal boron concentrations)
BV	Abnormal boron concentration: process system control rod has an abnormal boron concentration from burnup, dilution, or overaddition
BW	Overspeed: speed in excess of design limits
BX	Cladding failure: cladding of a component fails (e.g., the cladding of a fuel pellet is breached, and radioactive fuel leaks out)
BY	Burning/smoking: component is on fire or smoking
BZ	Engaged: component engages or meshes (this is not to be used when a component binds or becomes stuck or jammed)
CA	Disengaged/uncoupled: component disengages, loses required friction, or is no longer meshed (as in gears); for example, the clutch on the motor disengages from the shaft (this should not be used for dropped control rods)

Table 1.6 (continued)

Electric/instruments

EA Excessive electrical loads: electrical loads exceed design rating

EB Overvoltage/undercurrent: component failure produces an overvoltage/undercurrent condition other than open circuits

EC Undervoltage/overcurrent: component failure produces an undervoltage/overcurrent condition other than shorts

ED Short circuit/arcing/low impedance: electrical component shorts or arcs in the circuit or has a low impedance including shorts to ground

EE Open circuit/high impedance/bad electrical contact: electrical component has a structural break, or electrical contacts fail to contact and fail to pass the desired current

EF Erratic operation: component (especially electrical or instrument) behaves erratically or inconsistently (if an instrument produces a bad but constant signal, use "EG;" if an instrument produces an inconsistent signal use "EF")

EG Erroneous/no signal: electrical component or instrument produces an erroneous signal or gives no signal at all (not for out-of-calibration error)

EH Drift: a change in a setting caused by aging or change of physical characteristics (does not include personnel errors or a physical shift of a component)

EI Out of calibration: component (particularly instruments) become out of adjustment or calibration (does not include drift)

EJ Electromagnetic interference: abnormal indication or action resulting from unanticipated electromagnetic field

EK Instrument snubbing: dampening of pulsating signals to an instrument

Hydraulic

HA High flow: higher than normal or desired flow exists in a component/system (does not include instrument misindication (see code EG)

HB Low flow: lower than normal or desired flow exists in a component/system (does not include instrument misindication)

HC No flow or impulse: fluid flowing through a pipe, filter, orifice, or trench or the fluid in an impulse line (e.g., instrument sensing line) is blocked completely or decreased due to some foreign material, crud, closed (either partially or completely) valve or damper, or insufficient flow area

Table 1.6 (continued)

HD	Flow induced vibration
HE	Cavitation
HF	Erosion
HG	Vortex formation
HH	Water hammer
HI	Pressure pulse/surge
HJ	Air/steam binding
HK	Loss of pump section
HL	Boron precipitation
	<u>Other</u>
OA	Declared inoperable: component or system is declared inoperable as required by Technical Specifications but may be capable of partially or completely performing its desired duties when requested (a component/system that is <u>completely</u> failed should not use this code)
OB	Flux anomaly: flux characteristics of the reactor core are not as required or desired (e.g., flux spike due to xenon burnout)
OC	Test not performed: operator or test personnel fails to perform a required test within the required period
OD	Radioactivity contamination: component, system, or area becomes more radioactive than desired or expected
OE	Temporary modification: an installation intended for short term use (usually this is for maintenance or modification of installed equipment)
OF	Environmental anomaly
OG	Airborne release
OH	Waterborne release
OI	Operator communication
OJ	Operator incorrect action
OK	Procedure or record error

2. SOURCES OF INFORMATION

Several sources of information including periodic (annual, quarterly, and monthly) NRC publications were used in the review. Some sources contained information relative to more than one area within the scope of the review.

2.1 Availability and Capacity Factors

The availability and capacity factors were either extracted or calculated from data given in the Gray Books[3] from 1974 through 1981 (the first Gray Book was issued in May 1974). Prior to 1974, annual or semiannual reports were used to compile availability factors only.

2.2 Forced Reactor Shutdowns and Power Reductions

Review of the forced power reductions involved checking the following sources for accuracy and completeness of details.

1. Nuclear Power Plant Operating Experience for 19XX, for the years 1973–1979 (Refs. 4–11). The report for 1981 has not been published. However, because work on the section on outages in these reports has been performed by NSIC since 1973, the draft copy of this report for 1981 was available.
2. NUREG-0020 series[3] (Gray Books).
3. Annual or semiannual reports of the Dresden 2 plant from the time of startup through 1977. For 1977 through 1981, monthly operating reports were used because the utilities were no longer required to file annual reports. The review of power reductions involved primarily the annual, semiannual, and monthly reports.

2.3 Reportable Events

The NSIC computer file of LERs was the primary source of information in reviewing reportable events. Material on the NSIC computer file consists of the appropriate bibliographic material, title, 100-word abstract, and keywords. When additional information on the event was needed, the original LER (or equivalent) was consulted by examining (1) those full-sized copies on file at NSIC (for the years 1976–1981); (2) the microfiche file of docket material at NSIC; or (3) the appropriate operating report (semiannual, annual, or monthly).

Two computer files on RECON (a computer retrieval system containing ~40 data bases operated at ORNL) were used extensively. Printouts were obtained from the files for Dresden 2 to provide coverage on many types of "docket material," including reportable events, where the licensee may have been in correspondence with NRC [or the Atomic Energy Commission (AEC)] concerning a particular event. Licensees are often requested to submit additional information or perform further analysis. Before the

LERs came into existence in the mid-1970s, it was not unusual for licensees to submit, on their own or at the request of NRC or AEC, more than one letter transmitting information on a particular event. Thus, these printouts provided additional sources of information on reportable events.

Several special publications were reviewed to provide details on events of significance. After further analyses and examination of the following publications, details, evaluations, or assessments could be found other than those provided in the appropriate NRC-requested transmission.

1. "Reports to Congress on Abnormal Occurrences," NUREG-0090 series[12],
2. "Power Reactor Event Series" (formerly Current Event Series) published bimonthly by NRC,
3. "Operating Experiences," a section of each issue of the *Nuclear Safety* journal, and
4. the publications of NRC's Office of Inspection and Enforcement (IE), such as operating experience bulletins, IE bulletions, IE circulars, and IE information notices.

2.4 Environmental Events and Releases of Radioactivity

Events of environmental importance were obtained as a result of conducting the overall review of the plant's operating history, and the sources of information involve all types of documents listed thus far.

The data for radioactivity releases were compiled primarily from *Radioactive Materials Released from Nuclear Power Plants — Annual Report 1977* (Ref. 13). This report presents year-by-year comparisons for plants in a number of different categories (such as solid, gas, liquid, noble gas, and tritium). Data for 1978 were taken from *Radioactive Materials Released from Nuclear Power Plants — Annual Report 1978* (Ref. 14). Data for 1979, 1980, and 1981 were compiled from the annual environmental reports submitted by Dresden 2.

3. TECHNICAL APPROACH FOR EVALUATIONS OF OPERATING HISTORY

Forced shutdowns (and power reductions) and reportable events were the two areas focused on in the evaluation of the operating history of Dresden 2. Given the large number of both forced shutdowns and reportable events, it was necessary to develop consistent review procedures that involved screening and categorizing of both occurrences. After the events were screened and categorized, the study then assessed the safety significance of the events and analyzed the categories of events for various trends and recurring problems.

The approach in evaluation of operational events (forced shutdowns and reportable occurrences) consisted primarily of a three-step process: (1) compilation of information on the events, (2) screening of the events for significance using selected criteria and guidelines, and (3) evaluation of the significance and importance of the events from a safety standpoint. The evaluations were to determine those areas where safety problems existed in terms of systems, equipment, procedures, and human error.

Shutdowns were evaluated against the DBEs found in Chap. 15 of the *Standard Review Plan*.[2] The DBEs are those postulated disturbances in process variables or postulated malfunctions or failures of equipment that the plants are designed to withstand and that licensees analyze and include in safety analysis reports (SARs). The SAR provides the opportunity for the effects of anticipated process disturbances and postulated component failures to be examined to determine their consequences and to evaluate the capability built into the plant to control or accommodate such failures and situations (or to identify the limitations of expected performance).

The intent is to organize the transients and accidents considered by the licensee and presented in the SAR in a manner that will:

1. ensure that a sufficiently broad spectrum of initiating events has been considered,
2. categorize the initiating events by type and expected frequency of occurrence so that only the limiting cases in each group need to be quantitatively analyzed, and
3. permit the consistent application of specific acceptance criteria for each postulated initiating event.

Each postulated initiating event is to be assigned to one of the following categories:

1. increase in heat removal by the turbine plant,
2. decrease in heat removal by the turbine plant,
3. decrease in reactor coolant system flow rate,
4. anomalies in reactivity and power distribution,
5. increase in reactor coolant inventory,
6. decrease in reactor coolant inventory,
7. radioactive release from a subsystem or component, or
8. anticipated transients without scram.

Those shutdowns identified as design-basis initiating events were categorized as such. If the shutdown was not a DBE, then it was assigned a category from a list developed by NSIC to indicate the nature and type of error or failure. The NSIC categories for shutdowns not caused by DBEs were examined as part of a trends analysis.

Reportable events were screened using the criteria presented in Sect. 3.2 and were categorized according to their significance. The information collected on the reportable events was used to analyze trends for all reportable events, both significant and not significant.

3.1 Significant Shutdowns and Power Reductions

For the purposes of compiling information and evaluation, power reductions were treated in the same manner as forced shutdowns.

3.1.1 Criteria for significant shutdowns and power reductions

As indicated previously, the occurrences identified as DBEs were used as criteria to categorize and note significant shutdowns. These events are listed in Table 3.1 at the end of Sect. 3 as they are found in Chap. 15 of the *Standard Review Plan*.[2]

3.1.2 Use of criteria for determining significant shutdowns and power reductions

Generic design-basis initiating events such as "increase in heat removal by the secondary system" or "decrease in reactor coolant system flow rate," were used as primary flags for reviewing the forced shutdowns (and power reductions). Once the generic type of event was identified, the particular initiating event was determined from the details associated with the shutdown. For example, if the reactor shuts down because of an increase in heat removal because a feedwater regulator valve failed open, the shutdown is a generic type 1 DBE. Specifically, based on the initiating event (valve failed open), it is a 1.2 DBE — "feedwater system malfunction that results in an increase in feedwater flow." Some shutdowns were readily identifiable as specific DBEs, such as tripping of a main coolant pump, a 3.1 DBE. Once categorized as a DBE, the shutdown was considered significant regardless of the resulting effect on the plant (because a DBE had been initiated).

Loss of flow from one feedwater loop was considered sufficient to qualify as a 2.7 DBE — "loss of normal feedwater flow." The closure of a main steam isolation valve in one loop was considered sufficient to qualify as a 2.4 DBE — "inadvertent closure of main steam isolation valves."

3.1.3 Non-DBE shutdown and power reduction categorization

Those shutdowns that were not DBEs were assigned NSIC categories (Table 3.2) to provide more information on the failure or error associated with the shutdown. With these categories, more specific types of errors and failures could be examined through tabular summaries to focus the reviewer's attention on problem areas (safety related or not) that were not revealed by the DBE categories.

The causes (Table 1.1) for non-DBE shutdowns taken from the Gray Books are limited and very general, while NSIC cause categories are more specific. Thus, as an example, the number of Gray Book causes noted as equipment failure should not be expected to equal those identified as equipment failures with the NSIC categories. Other NSIC categories, such as component failure, could be classified as an equipment failure if the only available designations for cause were those listed in the Gray Books.

3.2 Significant Reportable Events

3.2.1 Criteria for significant reportable events

Two groups of criteria were used in determining significant reportable events. The first set of criteria (Table 3.3) indicates those events that are definitely significant in terms of safety; they are termed significant. The second set of criteria (Table 3.4) indicates events that may be of potential concern. These events, which might require additional information or evaluation to determine their full implication, were noted as conditionally significant.

3.2.2 Use of criteria for determining significant reportable events

The reportable events were all reviewed, applying the two sets of criteria for significance rather liberally. A number of significant events and conditionally significant events were noted. The events initially identified as significant or conditionally significant were analyzed and evaluated further based on (1) engineering judgment; (2) the systems, equipment, or components involved; or (3) whether the safety of the plant was compromised. The final evaluation for significance considered whether a DBE was initiated or whether a safety function was compromised so that the system as designed could not mitigate the progression of events. Thus, the number of events finally categorized as significant was reduced considerably by these steps in the review process.

3.2.3 Reportable events that were not significant

Those reportable events not identified as significant or conditionally significant were categorized as not significant (with an "N" in the significance column of the coding sheets in the appendixes). These events and the events rejected during the additional review step were further reviewed by compiling a tabular summary of the systems to detect trends and recurring problems (Table 1.4 provides a listing of the systems).

Table 3.1. Initiating event descriptions for DBEs as listed in Chap. 15, Standard Review Plan (Revision 3)

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1. Increase in heat removal by the secondary system
   - 1.1 Feedwater system malfunction that results in a decrease in feedwater temperature
   - 1.2 Feedwater system malfunction that results in an increase in feedwater flow
   - 1.3 Steam pressure regulator malfunction or failure that results in increasing steam flow
   - 1.4 Inadvertent opening of a steam generator relief or safety valve
   - 1.5 Spectrum of steam system piping failures inside and outside of containment in a pressurized-water reactor (PWR)
   - 1.6 Startup of idle recirculation pump[a]
   - 1.7 Inadvertent opening of bypass resulting in increase in steam flow[a]
2. Decrease in heat removal by the secondary system
   - 2.1 Steam pressure regulator malfunction or failure that results in decreasing steam flow
   - 2.2 Loss of external electric load
   - 2.3 Turbine trip (stop valve closure)
   - 2.4 Inadvertent closure of main steam isolation valves
   - 2.5 Loss of condenser vacuum
   - 2.6 Coincident loss of onsite and external (offsite) ac power to the station
   - 2.7 Loss of normal feedwater flow
   - 2.8 Feedwater piping break
   - 2.9 Feedwater system malfunctions that result in an increase in feedwater temperature[a]
3. Decrease in reactor coolant system flow rate
   - 3.1 Single and multiple reactor coolant pump trips
   - 3.2 Boiling-water reactor (BWR) recirculation loop controller malfunction that results in decreasing flow rate
   - 3.3 Reactor coolant pump shaft seizure
   - 3.4 Reactor coolant pump shaft break
4. Reactivity and power distribution anomalies
   - 4.1 Uncontrolled control rod assembly withdrawal from a subcritical or low-power start-up condition (assuming the most unfavorable reactivity conditions of the core and reactor coolant system), including control rod or temporary control device removal error during refueling
   - 4.2 Uncontrolled control rod assembly withdrawal at the particular power level (assuming the most unfavorable reactivity conditions of the core and reactor coolant system) that yields the most severe results (low power to full power)
   - 4.3 Control rod maloperation (system malfunction or operator error), including maloperation of part length control rods

Table 3.1 (continued)

4.4 Start-up of an inactive reactor coolant loop or recirculating loop at an incorrect temperature.

4.5 A malfunction or failure of the flow controller in a BWR loop that results in an increased reactor coolant flow rate

4.6 Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant of a PWR

4.7 Inadvertent loading and operation of a fuel assembly in an improper position

4.8 Spectrum of rod ejection accidents in a PWR

4.9 Spectrum of rod drop accidents in a BWR

5. <u>Increase in reactor coolant inventory</u>

5.1 Inadvertent operation of emergency core cooling system during power operation.

5.2 Chemical and volume control system malfunction (or operator error) that increases reactor coolant inventory

5.3 A number of BWR transients, including items 1.2 and 2.1–2.6

6. <u>Decrease in reactor coolant inventory</u>

6.1 Inadvertent opening of a pressurizer safety or relief valve in either a PWR or a BWR

6.2 Break in instrument line or other lines from reactor coolant pressure boundary that penetrate containment

6.3 Steam generator tube failure

6.4 Spectrum of BWR steam system piping failures outside of containment

6.5 Loss-of-coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary, including steam line breaks inside of containment in a BWR

6.6 A number of BWR transients, including items 1.3, 2.7, and 2.8

7. <u>Radioactive release from a subsystem or component</u>

7.1 Radioactive gas waste system leak or failure

7.2 Radioactive liquid waste system leak or failure

7.3 Postulated radioactive releases due to liquid tank failures

7.4 Design basis fuel handling accidents in the containment and spent fuel storage buildings

7.5 Spent fuel cask drop accidents

8. <u>Anticipated transients without scram</u>

8.1 Inadvertent control rod withdrawal

8.2 Loss of feedwater

8.3 Loss of ac power

8.4 Loss of electrical load

8.5 Loss of condenser vacuum

8.6 Turbine trip

8.7 Closure of main steam line isolation valves

[a]These initiating events were added for BWRs to be more specific than DBE events 5.3 and 6.6.

Table 3.2. NSIC event categories for non-DBE shutdowns

N 1.0	Equipment failure
	N 1.1 Failure on demand under operating conditions
	N 1.1.1 Design error
	N 1.1.2 Fabrication error
	N 1.1.3 Installation error
	N 1.1.4 End of design life/inherent failure/random failure
	N 1.2 Failure on demand under test conditions
	N 1.2.1 Design error
	N 1.2.2 Fabrication error
	N 1.2.3 Installation error
	N 1.2.4 End of design life/inherent failure/random failure
N 2.0	Instrumentation and control anomalies
	N 2.1 Hardware failure
	N 2.2 Power supply problem
	N 2.3 Setpoint drift
	N 2.4 Spurious signal
	N 2.5 Design inadequacy (system required to function outside design specifications
N 3.0	Non-DBE reductions in coolant inventory (leaks)
	N 3.1 In primary system
	N 3.2 In secondary system and auxiliaries
N 4.0	Fuel/cladding failure (densification, swelling, failed fuel elements as indicated by elevated coolant activity)
N 5.0	Maintenance error
	N 5.1 Failure to repair component/equipment/system
	N 5.2 Calibration error
N 6.0	Operator error
	N 6.1 Incorrect action (based on correct understanding on the part of the operator and proper procedures, the operator turned the wrong switch or valve — incorrect action)
	N 6.2 Action on misunderstanding (based on proper procedures and improper understanding or misinterpretation on the operator's part of what was to be done — incorrect action)
	N 6.3 Inadvertent action (purpose and action not related, for example, bumping against a switch or instrument cabinet)
N 7.0	Procedural/administrative error (incorrect operating or testing procedures, incorrect analysis of an event — failure to consider certain conditions in analysis)
N 8.0	Regulatory restriction
	N 8.1 Notice of generic event
	N 8.2 Notice of violation
	N 8.3 Backfit/reanalysis

Table 3.2 (continued)

N 9.0	External events	
	N 9.1	Human induced (sabotage, plane crashes into transformer)
	N 9.2	Environment induced (tornado, severe weather, floods, earthquake)
N 10.0	Environmental operating constraint as set forth in Technical Specifications	

Table 3.3. Reportable event criteria — significant

Category of significance	Event description
S1	Two or more failures occur in redundant systems during the same event
S2	Two or more failures due to a common cause occur during the same event
S3	Three or more failures occur during the same event
S4	Component failures occur that would have easily escaped detection by testing or examination
S5	An event proceeds in a way significantly different from what would be expected
S6	An event or operating condition occurs that is not enveloped by the plant design bases
S7	An event occurs that could have been a greater threat to plant safety with (1) different plant conditions, (2) the advent of another credible occurrence, or (3) a different progression of occurrences
S8	Administrative, procedural, or operational errors are committed that resulted from a fundamental misunderstanding of plant performance or safety requirements
S9	Other (explain)

Table 3.4. Reportable event criteria — conditionally significant

Category of conditional significance	Event description
C1	A single failure occurs in a nonredundant system
C2	Two apparently unrelated failures occur during the same event
C3	A problem results in an offsite radiation release or exposure to personnel
C4	A design or manufacturing deficiency is identified as the cause of a failure or potential failure
C5	A problem results in a long outage or major equipment damage
C6	An engineering safety feature actuation occurs during an event
C7	A particular occurrence is recognized as having a significant recurrence rate
C8	Other (explain)

4. OPERATING EXPERIENCE REVIEW OF DRESDEN 2

4.1 Summary of Operational Events of Safety Importance

The operational history of Dresden 2 has been reviewed to indicate those areas of plant performance that have compromised plant safety. The review included a detailed examination of plant shutdowns, power reductions, reportable events, and special environmental impacts. The criteria used to show degradations in plant safety were (1) events that initiated a DBE and (2) events that compromised safety functions designed to mitigate the propagation of the initiating events.

Shutdowns and power reductions indicated the number and types of DBEs entered. The reportable events and special environmental impacts indicated the number of times each engineered safety function was compromised. The results of the analyses identified 68 DBEs entered. Additionally, four events were identified where loss of safety system function occurred in engineered safety features.

4.2 General Plant Description

Dresden Unit 2 is a single cycle boiling water reactor (BWR) owned by Commonwealth Edison Company and located in Gooselake Township, Grundy County, Illinois. The nearest city, at a distance of 11 miles, is Davenport, Iowa. The population within 30 miles is 430,000 and is 670,000 within 50 miles. The net maximum dependable capacity is 794 MWe except during the period when plant capability is limited by the condenser water cooling facility's ambient temperature, normally July and August. During that period, net maximum dependable capacity is 772 MWe. The nuclear steam supply system is a BWR class 3 utilizing a Mark 1 containment and was manufactured by the General Electric Company. The architect engineer was Sargent and Lundy, and the constructor was United Engineers and Constructors. The condenser cooling method is of the once through type with the source of cooling water being the Dresden 2 unit cooling lake with use of the Kankakee River in accordance with the limitations specified by NPDES Permit No. IL0002224. The plant is subject to License No. DPR-19, issued December 22, 1969, pursuant to Docket No. 50-237. The date of initial reactor criticality was January 7, 1970, and commercial operation began June 9, 1972.

4.3 Availability and Capacity Factors

Table 4.1 contains the availability and capacity factors for Dresden 2. Dresden 2 began commercial operation on June 9, 1972. The reactor availability in the period from 1972 through 1981 was above 70% except for

Table 4.1 Dresden 2 availability and capacity factors

	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	Cumulative
Reactor availability	44.5	69.3	62.5	90.8	66.8	57.8	78.3	73.5	96.3	83.2	95.6	62.8	77.1
Unit availability	34.0	65.0	59.8	87.6	64.1	55.1	75.9	71.9	94.2	81.6	93.3	60.1	73.3
Unit capacity (MDC)[a]	ND[b]	ND	ND	ND	ND	42.3	64.5	52.2	84.4	73.0	67.6	50.4	57.0
Unit capacity (DER)[c]	22.4	37.3	45.2	70.8	48.2	41.2	61.6	50.8	82.0	71.0	65.7	49.0	55.4

[a]MDC = maximum dependable capacity.
[b]ND = no data.
[c]DER = design electrical rating.

four years: 1972, 1974, 1975, and 1981. Leaks, pipe cracks, valve failures, and other problems accounted for lower availabilities in these years. In the ten years of commercial operation, the reactor availability has averaged 77.1%. Unit availability and unit capacity (DER) during this ten-year period have averaged 73.3% and 57.0%, respectively. Unit capacity (MDC) was not available in the years prior to 1975. During the period of 1975 through 1981, unit capacity (MDC) averaged 55.4%.

4.4 Review of Forced Reactor Shutdowns and Forced Power Reductions

Tables A.1.1 through A.1.12 in Appendix A provide a comprehensive summary of information concerning shutdown and power reductions at Dresden 2. Tables 4.2 and 4.3 summarize Tables A.1.1 through A.1.12 for forced shutdowns and power reductions, respectively. The duration of the event is rounded to the nearest hour for a forced outage or zero for a power reduction. The power (%) at the beginning of the event was approximated from the average daily power charts if no specific value was available.

There were 188 forced outages and 18 reported power reductions for the twelve-year reporting period 1970 through 1981 with an average of 16 forced outages per year. The average time the plant was shut down due to these occurrences was 893 h/year. Approximately one-third of the occurrences were identified as DBE-initiating events as defined in Sect. 3.

4.4.1 Yearly summaries for Dresden 2

1970

On January 7, 1970, at 0122, the Dresden 2 BWR achieved initial criticality and startup power testing began. The generator was first synchronized to the grid on April 13. Approximately 20% of the total number of 37 forced outages/power reductions during the review period covered (1970–1981) occurred the first year. These accounted for 2890 h or 27% of the total hours of forced downtime. Eleven of these 37 events are events that one would normally expect during the very early operation. Most of these shutdowns were associated with the reactor trip system. Six of the 37 events represented unnecessary challenges to the reactor protection and safety systems where instrument technicians, mechanics, and maintenance personnel introduced false or spurious signals. Most of these events involved the feedwater system.

A significant event occurred on June 5 involving an unscheduled blowdown. This outage lasted 59 days and accounted for one-half of the forced outage time for 1970. More details on this event are in the section on reportable events (Sect. 4.5.2.3).

Other forced outages/power reductions of interest were:

March 30: First condenser tube leak encountered accounting for 137 h forced downtime.

Table 4.2. Dresden 2 forced shutdown summary

	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	Total
I. Forced shutdowns													
1. Total number	37	20	21	18	12	14	12	9	12	10	13	10	188
2. Total hours down	2890	486	1292	808	1508	588	438	706	512	470	588	432	10718
3. Cause [a]													
A. Equipment failure	22(2586)	5(56)	13(433)	7(108)	10(1165)	8(276)	8(158)	5(135)	7(378)	5(190)	8(464)	9(429)	107(6378)
B. Maintenance or testing	6(216)	9(371)	6(854)	6(355)		4(249)	3(159)	2(533)		1(54)	1(8)		38(2799)
D. Regulatory restriction				2(332)						1(183)	1(63)		4(578)
E. Operator Training/ License exam													
F. Administrative									1(27)				1(27)
G. Operational error	9(88)	6(59)	2(5)	3(13)	2(343)	2(63)	1(121)	2(38)	2(60)	3(43)	3(53)		35(886)
H. Other									2(47)			1(3)	3(50)
4. Shutdown method													
1. Manual	8	8	10	9	10	10	2	3	4	6	1	4	74
2. Manual scram	3					1	3				5	1	13
3. Automatic scram	26	12	11	9	2	3	5	5	8	4	7	4	96
4. Other							2	1				1	4
II. Total number of DBE related shutdowns (These are included in Totals of Part I)	19(2074)	7(48)	8(64)	6(41)	1(12)	1(66)	5(108)	2(39)	5(98)	2(27)	5(274)	3(124)	64(2975)

Table 4.2. (Cont.)

	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	Total
III. System involved													
1. Reactivity control system (RB)	3	1	1	1	1						3		10
2. Coolant recirculation systems and controls (CB)	2	1	2	3	3		1	3	2		1	4	22
3. Main stream systems and controls (CC)	2	6		1	3	2	1				1	1	17
4. Main steam isolation system and controls (CD)	2	2			1				1	1			7
5. Feedwater systems and controls (CH)	6	1		5	1		2			1		1	17
6. Containment combustible gas control systems and controls (SE)						2							2
7. High pressure safety injection system (SF-C)	2		1										3
8. Core spray system (SF-D)										1			1
9. Other engineered safety feature systems and controls (SH)				2		1				1			4
10. Reactor trip systems (IA)	9	1	1			1		2		1	3	1	19
11. Other instrument systems required for safety (IE)							1		1	2			4

Table 4.2. (Cont.)

	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	Total
12. Offsite power (EA)									1				1
13. AC onsite power (EB)	3						1	1	1				6
14. Compressed air systems (PA)	1	4	1	1		1		1	1				10
15. Chemical, volume control, and liquid poison systems (PC)							1						1
16. Air conditioning, heating, cooling, and ventilation systems (AA)									1	1			2
17. Turbine-generators and controls (HA)	3	2	10	4	3	7	3		3		1	2	39
18. Main-steam supply system (HB)			3				1	1		2			7
19. Main condenser systems (HC)	4	2	2				1		1		2		12
20. Gaseous radioactive waste management systems (MB)				1							1		2
21. Unknown								1					1
22. Onsite systems and controls (ED)												1	1

[a]Number of hours associated with cause of shutdown is in parentheses.

Table 4.3 Dresden 2 forced power reduction summary

	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	Total
I. Power reductions													
1. Total number					0	1	4	7	2	4			18
2. Cause													
A. Equipment failure						1	2	7	1	4			15
B. Maintenance or testing							1		1				2
D. Regulatory restriction													
E. Operator training/license exam													
F. Administrative													
G. Operational error							1						1
H. Other													
3. System involved													
1. Coolant recirculation (CB)							1	5	1	3			10
2. Main steam isolation (CD)									1				1
3. Feedwater (CH)							2						2
4. Turbine-generators (HA)						1				1			2
5. Main steam supply (HB)								1					1
6. Main condenser (HC)							1	1					2
4. Total number of DBEs related power reductions (included in totals of Part I)							2	1		1			4

May 14: Both feedwater pumps tripped resulting in a total loss of feedwater.

October 13: Another condenser tube leak accounting for 244 h of forced downtime this year.

November 14: Both reactor feedwater pumps repaired and five control rod drives (CRDs) replaced accounting for 110 h of forced downtime.

December 7: After trying to repair the main transformer twice on September 23 and September 24, the main transformer finally failed, and required 465 h to replace.

1971

The startup power test program was finally completed on January 28 when the unit load was increased to full power. The first refueling outage began February 26. There were 20 forced outages accounting for 486 h of forced downtime. There were six events involving operator error which are be discussed as part of the trends analysis in Sect. 4.4.4.

Several new problems were encountered this year. First, defective fuel and off-gas release rates prevented the unit from operating at full power. Second, 20% of the forced shutdowns for 1971 accounting for 170 h of forced downtime were due to moisture separator problems in the turbine. Most of the trips associated with the moisture separator problems were due to spurious levels signals. It was finally determined that vibration was causing the mercury switches in the moisture separator level sensor to trip. These switches were replaced with snap-action switches. A time-delay was also installed into the trip circuit. Very few problems with this system have been encountered since then.

Problems with compressed air to the valve operators of main steam isolation valves (MSIV) accounted for 20% of the shutdowns and for 93 h of forced downtime. The MSIVs are designed to fail closed on loss of pneumatic pressure to the valve operator. This problem continued to plague Dresden 2 throughout its operating history.

Forced outages/power reductions of interest are:

June 7: Shutdown for 122 h to modify the turbine moisture separator baffle.

June 14: Shutdown to investigate high levels occurring in the scram discharge volume which would have the potential to restrict the capability of the reactor to scram.

July 20: A partial scram occurred during testing of main steam line radiation monitors. A load reduction and pressure transient followed. This event is typical of several forced outages at Dresden 2 in that they occurred during some type of testing procedure and are discussed as part of the trends analysis in Sect. 4.4.4.

1972

The second refueling outage began February 19. Dresden 2 began commercial operation on June 9. There were 21 forced outages accounting for 1292 h of downtime during 1972. The majority of this forced downtime (853 h or 66%) was due to flow restrictor problems in the main steam line.

This event is discussed in more detail in the reportable events section (Sect. 4.5.1.2.2). Two new problems surfaced this year which have recured in subsequent years. The first problem concerned oil leaks in the turbine electrohydraulic control (EHC) system. One-fifth of the outages this year were due to these problems. The second problem concerned oil and seal leaks in the recirculation pumps. These accounted for 292 h of the forced downtime.

Other shutdowns of interest are:

June 12: Third major outage due to condenser tube leaks lasting only 42 h.

July 17: Recirculation pump seal leaks accounted for 98 h of downtime.

September 7 and October 14: Main steam line flow restrictor problem and repair accounted for 853 h (66%) of downtime.

December 22: Recirculation pump seals leak accounted for 194 h of downtime.

1973

There were 18 forced outages accounting for 808 h of downtime. The unit was shut down on March 25 for modification of the off-gas system for 211 h (26%). During this outage, an explosion occurred in the off-gas system when a welder came across an unvented hydrogen bubble in the system. Regulatory requirements forced the unit to shut down on August 2 and October 5 for inspection of snubbers for a total of 332 h (41%) of downtime.

A new problem also occurred during this year. Low reactor water level occurred four times as a result of problems with feedwater regulating valves. This has subsequently become a recurring problem. Three outages were due to oil or seal leaks on the recirculation pumps. Again, another outage was due to problems associated with the compressed air supply to a MSIV.

1974

There were twelve forced shutdowns, yet the downtime was 1508 h which was the second highest total from the standpoint of forced outages in Dresden's operating history.

On February 12, the unit shut down for 144 h to replace a feedwater check valve seal and two recirculation pump seals. On February 18, while bringing the unit back on line, the turbine generator turning gear was damaged requiring 329 h to repair. On August 22, problems were experienced with an uncoupling of a couple of control rods resulting in 112 h of downtime. On September 12, cracks were found in recirculation system piping. The unit remained shut down for 24 d correcting this problem. A more thorough discussion is provided in the reportable events section (Sect. 4.5.2.1) on the cracks in the recirculation system piping. On October 19, a bad recirculation pump seal had to be replaced. The unit was down 99 h. The third refueling outage began in early November. Thus, forced outages due to recirculation pump seals, oil leaks, and feedwater valves continued to recur.

1975

The year commenced with a continuation of the refueling outage began the year before. There were 14 forced outages accounting for 588 h downtime and one power reduction. On May 23 the unit was shut down for 66 h due to main steam isolation valve closure resulting from a faulty switch. The internals of the switch had inadvertently been left out and the switch was repaired. In connection with this shutdown, the unit was having problems with electromatic relief valves. The valves failed to open due to the mechanical operating arms having too much clearance between cap screw and pilot valve steam.

On June 13 the unit again shut down for 85 h due to problems with two electromatic relief valves when the valves failed to operate. They were overhauled and cleaned, new rings were installed, and they were put back into service. On September 24, another required inspection of snubbers took place resulting in 93 h downtime. On October 8, cracks were found in the nitrogen inerting piping with repairs requiring 126 h. Almost one-third of the outages for 1975 were due to oil leaks in the turbine EHC system.

1976

Dresden 2 experienced 12 forced shutdowns totaling 438 h and four forced power reductions during 1976. The fourth refueling outage began March 14. During the refueling outage, fatigue failure of welds on most of the jet pump assemblies was discovered and repaired. The unit shut down on May 19, when an accidental boron injection occurred while starting up shortly after the refueling outage was completed. The unit remained shut down for 121 h. On June 25 and November 13, maintenance and repair were performed on the TIP machines accounting for 135 h of downtime. Two outages were required to repair turbine EHC oil leaks, one outage to repair turbine control valves, one outage and two power reductions to repair recirculating pump oil and seal leaks, and two outages to perform maintenance on feedwater regulating valves.

1977

The fifth refueling outage was conducted this year beginning September 10. From November 23, this outage was declared to be a forced maintenance outage since refueling had been completed, but maintenance had not been completed. This forced outage accounted for 428 h of the year's total downtime of 706 h. There were only nine forced outages while there were seven forced power reductions. Recirculating pump seal and oil problems accounted for two of the outages and four of the power reductions.

1978

For 1978, 12 forced outages required 512 h of downtime, and two power reductions were required. On May 20, the unit went down due to condenser tube leakage and remained down for 76 h. On August 1, a bonnet leak on one of the recirculation pumps forced the unit to remain down for 179 h. The unit shut down for 27 h on June 25 because of a generator load rejection due to a phase mismatch during a severe storm. On July 28, a contractor crew dug into an instrument air line leading to the filter building. This shut down the reactor for 20 h. On July 29, an operator, by using a walkie-talkie in the auxiliary electric room, caused the turbine 12% overspeed circuit to trip. One forced shutdown and one power reduction were attributable to brush replacements on a recirculation pump motor/generator (M/G) set and to a trip of the lubrication oil pump, respectively.

1979

The sixth refueling outage began March 17. During 1979, there were ten forced outages requiring the unit to be shut down for 470 h, and four power reductions. The only major outage involved a required inspection of snubbers on October 13 for 183 h. TIP machines required maintenance twice. Two power reductions were due to problems with recirculation pumps. There was one inadvertent closure of an MSIV. Two of the outages were due to operator errors. The first involved the feedwater system, and the second the reactor trip system. These are discussed as part of the trends analysis in Sect. 4.4.4.

1980

Thirteen forced outages totaling 588 h occurred in 1980. On May 12, the unit went down for 145 h to replace recirculation pump seals. On July 26, the unit was required to shut down to verify the CRD system accounting for 63 h. On October 9, an EHC pump malfunctioned. The unit was down 84 h because of HPCI problems preventing startup. On September 23, leakage from turbine hood accounted for 78 h downtime while moisture in the turbine vibration meter required the unit to shut down on December 2 for 73 h. There was one forced outage attributable to recirculation pump problems. There were three outages due to operational error: one dealt with the feedwater instrument rack while the other two involved inadvertent closure of MSIVs.

1981

Dresden 2 experienced 10 forced shutdowns in 1981 resulting in a total of 432 h of downtime. The turbine was manually tripped due to a generator ground on May 12 and then was down for 12 h. On June 13 a lightning strike created electrical disturbances resulting in an automatic scram and subsequent shutdown for 50 h. Problems were encountered with recirculation pump 2B on June 30, July 15, and August 15. The June 30 event involved a manual trip of the reactor after the recirculation pump tripped while an operator was adjusting the flow. On July 15, the reactor was manually scrammed to perform repairs on the recirculation pump 2B. The third recirculation pump event required a shutdown to perform repairs on the 2B recirculation oil pump. Mechanical problems were encountered on September 21 with the circuitry of the turbine stop valves. The unit was shut down for 99 h on November 3 to replace a recirculation pump seal. Two low reactor water level scrams were experienced on December 3 and 12, resulting in shutdowns of 17 and 75 h, respectively. The final shutdown for 1981 occurred on December 24. While performing safety relief operability surveillance, 2B safety relief valve failed to open. The HPCI was inoperable and the reactor was shut down.

4.4.2 Forced shutdowns and forced power reductions caused by DBE initiating events

Through 1981, Dresden 2 experienced 188 forced shutdowns of which 64 were identifiable as DBE initiating events. Dresden 2 experienced 18 power reductions with 4 identifiable as DBE initiating events for a total of 68 (Table 4.4). The DBEs accounted for 2975 h of downtime (28% of total hours shut down) and for 34% of the total number of forced shutdowns. One DBE accounted for 1417 h or one-half of the hours due to DBE initiating events. Excluding this one event, the DBE initiating events accounted for one-third of the total number of shutdowns for a downtime of 1558 h.

4.4.2.1 DBE Category 1 — increase in heat removal. Eleven events were identified as DBE Section 1 events. Seven of the eleven were operational errors made by either operators, instrument technicians, mechanics, or maintenance personnel. Six of these seven involved the feedwater system.

4.4.2.1.1 D1.2—feedwater system malfunctions resulting in an increase in feedwater flow. Three (4/16/70, 5/21/70, and 5/28/70) of the eight events were caused by test engineers or instrument mechanics introducing a spurious signal into the system. Three additional events (9/11/70, 9/10/71, and 9/29/73) were due to flow or level spikes occurring as a result of operator error. On August 28, 1970, a feedwater control valve stuck open. On September 19, 1976, a blown fuse in the feedwater regulating valve control circuit resulted in increase in feedwater flow.

4.4.2.1.2 D1.6 — startup of idle recirculation pump. The one D1.6 event occurred on March 24, 1970 when an idle recirculation pump was started, an IRM trip occurred due to cold water addition resulting from the pump startup.

4.4.2.1.3 D1.7 — inadvertent opening of bypass valves resulting in an increase in steam flow. There were two events of this type. On April 27, 1970, a test engineer accidentally introduced a signal into a pressure regulator causing a step decrease in setpoint. This resulted in the bypass valves opening and an increase in steam flow. The second event occurred on June 5, 1970, which is reported in the reportable events section (Sect. 4.5.2.3).

Table 4.4 Dresden 2 DBE initiating event summary

	DBE Category	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	Total
Feedwater system malfunctions that result in an increase in feedwater flow	D1.2	5	1		1			1						8
Startup of idle recirculation pump	D1.6	1												1
Inadvertent opening of bypass valves resulting in an increase in steam flow	D1.7	2												2
Loss of external electric load	D2.2			3						1	1			5
Turbine trip	D2.3	4	4	3		1		1		4		2	1	20
Inadvertent closure of MSIVs	D2.4	1	1	2	1		1		1		1	1		9
Loss of condenser vacuum	D2.5	2	1					1	1			2		7
Loss of normal feedwater flow	D2.7	2			4			1	1		1		1	10
Single and multiple reactor coolant pump trips	D3.1							2					1	3
Control rod maloperation	D4.3	1												1
Inadvertent opening of safety relief valve	D6.1	1						1						2
Total		19	7	8	6	1	1	7	3	5	3	5	3	68

4.4.2.2 DBE category 2 — decrease in heat removal. Forty-nine events are categorized into this section. Thirteen of these 49 (27%) were connected with operational error by operators, instrument technicians, mechanics, or maintenance people. Four of these were associated with turbine trips, five with inadvertent MSIV closure, one with low condenser vacuum, one with feedwater system, one with control rod operation, and one with load rejection.

4.4.2.2.1 D2.2 — loss of external electric load. On June 22, 1972, EHC oil pressure oscillation occurred during turbine surveillance. This resulted in the low EHC oil pressure sensors at the turbine control valves causing the "load reject" relays to operate.

On August 6, 1972, low EHC oil pressure caused a load reject scram. On December 12, 1972, a spurious generator trip resulted from the loss of field. On June 25, 1979, a generator load rejection occurred due to phase mismatching during a severe storm. On June 6, 1979, a transformer in generator field control failed resulting in a load reduction.

Thus, two of the load rejections were attributable to low EHC oil pressure with one of these due to operational error.

4.4.2.2.2 D2.3 — turbine trip. Four of the twenty events (6-1-70, 7-2-71, 7-19-71, 7-29-78) are attributable to operational error which will be discussed later in Sect. 4.4.4 on trends. Six of these events (10-10-70, 6-7-71, 6-14-71, 8-20-72, 3-16-77, 6-27-76) were attributable to spurious moisture separator level signals. It was discovered that vibration was causing the mercury switches to trip, and they were replaced with snap action switches and a delay on the trip circuit which successfully corrected the problem.

On 5-23-70, a sensor failed causing stop valve closure. On 1-1-78, a faulty switch on a test circuit tripped the reactor from stop valve closure. Four of the events (6-16-72, 8-17-72, 2-9-78, 2-10-79) were referenced as stop valve closure with no other information provided. These are considered as spurious signals.

On 12-02-80, moisture in the turbine vibration meter caused a turbine trip on stop valve closure. On 10-9-80, an EHC electrical pump malfunctioned causing stop valve closure. On May 12, 1981, a generator ground resulted in manually tripping the turbine.

Of the twenty events, one can be considered as a major event from an economic point of view. In the fall of 1970, the main transformer was leaking oil. The unit shut down twice for repairs of the main transformer. Finally, on December 7, 1970, the main transformer failed completely causing a generator and turbine trip. The unit remained down 465 h.

4.4.2.2.3 D2.4 — inadvertent closure of main steam isolation valves. There were nine events categorized in this section. Three of the nine (5-19-70, 5-23-72, 11-24-80) were due to operational error during tests. Two of the nine events (5-23-75, 6-12-79) resulted from causes unknown or not reported. The remaining four events resulted from loss of air to the MSIV. Two of these were attributable to operational error. On 7-13-71, the drywell pneumatic air supply was being isolated for repairs.

The backup air supply was inadvertently isolated causing MSIV closure. On 6-25-72, a fitting on a test pilot was left without being replaced. It subsequently broke and allowed air to be bled off the valve operator which caused MSIV closure.

On October 19, 1973, a pneumatic air supply valve was closed resulting in MSIV closure. On September 7, 1977, a loss of air occurred due to valve diaphragm failure which closed the MSIV.

4.4.2.2.4 <u>D2.5 - loss of condenser vacuum.</u> While the classification of D2.5 deals with complete loss of condenser vacuum, it was felt that any events which dealt with low condenser vacuum should be included in this category. None of the events dealt with a complete loss of condenser vacuum. Of the seven events, five (5-8-70, 9-12-70, 6-14-71, 1-2-76, 6-14-77) were reported as reactor scram on low condenser vacuum. On February 3, 1980, low vacuum occurred while transferring maximum recycle boiler relief discharge for unit 3 condenser to unit 2 condenser. On September 23, 1980, excessive air leakage into the condenser from the turbine hood caused scram from low vacuum.

4.4.2.2.5 <u>D2.7 - loss of normal feedwater flow.</u> Of the ten events, four events (1-18-73, 2-19-73, 11-14-73, 2-18-76) were due to feedwater regulator valve malfunctions.

On May 14, 1970, both operating reactor feed pumps tripped on low suction pressure caused by high condensate demineralized differential pressure. The reactor was scrammed to minimize the transient.

On May 26, 1970, the loss of the essential service MG set caused a loss of the feedwater control system. The reactor scrammed on low water level.

On November 27, 1973, a feed pump trip caused low water level. While restarting the pump, bus 22 was lost, and thus both pumps were lost, resulting in reactor scram on low water level.

On June 4, 1977, when a high pressure trip switch was valved back into sensing line which is common to reactor low level trip switch, the unit scrammed on reactor water low level trip. On November 10, 1979, a feed-water pump tripped due to low suction pressure trip introduced by instrument mechanics.

On December 12, 1981, after repairing steam leaks in the X-area and placing the generator back on line, the reactor scrammed on low reactor water level. Because of the lack of information and the past feedwater problems, this is assumed to be a potential and highly probable loss of feedwater event.

4.4.2.3 <u>DBE category 3 - decrease in reactor coolant system flow rate.</u> On February 13, 1976, a recirculating pump valve packing leak forced the unit down. On October 17, 1976, the 2A1 lubrication oil pump tripped which resulted in the M/G set and recirculation pump being tripped. The other recirculation pump was set to minimum flow. The unit load was decreased to 50% power. The lubrication pump was restarted and load measure was commenced.

On June 30, 1981, recirculation pump "2B" tripped while an operator was adjusting flow. The reactor was manually shut down to repair the "2B" recirculator pump. Originally reported as operator error, the event was later reclassified as equipment failure. On July 15, August 15, and November 3, 1981, the reactor was shut down to do more maintenance work on this pump.

4.4.2.4 DBE category 4 — reactivity and power distribution anomalies. On April 6, 1970, the control rods were inserted in such a fashion that the cooldown rate was sufficient to add enough reactivity to cause the reactor to scram from IRM trips.

4.4.2.5 DBE category 6 — decrease in reactor coolant inventory. On November 19, 1970, one electromatic relief valve failed. The cause of malfunction was a shorted lamp socket on electromatic relief valve "open" light. On May 5, 1976, during startup, a target rock safety relief valve stuck open.

4.4.3 Non-DBE forced shutdowns

Table 4.5 summarizes the NSIC categories assigned to non-DBE shutdowns. Only the major NSIC categories are listed in Table 4.5. Equipment failures accounted for one-half of the events with an apparent decline after the first two or three years of operation. Although instrumentation and control problems accounted for 10% of the events, the majority of these were in the first three years. The human factors-related categories (5, 6, and 7) involved 15% of the events. Almost 20% of the events involved leaks in the coolant inventory system. Even with the large number of these events in 1974 when the pipe cracks were discovered, the cracks are appearing at the rate of two or three of these events per year.

The one external event involved a severe storm, and the reactor was shut down due to a generator load rejection as a result of a phase mismatch.

4.4.4 Trends and safety implications of forced shutdowns and forced power reductions

Several trends are observed as a result of this study of the forced outages and power reductions for Dresden 2. The trends deal with the following items: (1) operator error, (2) leaks and pipe cracks, (3) turbine control valves and turbine EHC oil leaks, (4) inadvertent MSIV closures, and (5) feedwater regulator valve problems.

4.4.4.1 Operator error. In considering operator error, the following categories naturally fall out: (1) surveillance testing in progress, (2) maintenance activities, (3) procedure problems, (4) incorrect action, and (5) inadvertent action. First, eleven the 36 events associated with operator error dealt with the unit while surveillance testing was in progress (4-11-70, 4-1670, 4-27-70, 5-21-70, 6-1-70, 9-11-70, 7-2-71, 7-8-75, 2-871, 11-10-79, 11-24-80). All but three of these events occurred before or during 1975. The general sequence of events was as follows: while surveillance testing was in progress, an instrument mechanic or a test engineer accidently introduced a spurious signal which resulted in a reactor trip.

Six of the events were associated with maintenance activities. Three (7-31-71, 7-19-71, 6-4-77) dealt with instrument or sensing lines. The

Table 4.5 Dresden 2 non-DBE initiating event summary

	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	Total
1.0 Equipment failures	8	8	7	6	3	10	5	3	3	4	3	3	63
2.0 Instrumentation and controls anomalies	7	2	2		1						1	2	15
3.0 Non-DBE reductions in coolant inventory (leaks)	2	1	3	2	5	1	1	3	1	2	1	1	23
4.0 Fuel/cladding failure													
5.0 Maintenance error													
6.0 Operator error	1	2	1	2	2	2		1	1	1	2		15
7.0 Procedural/administrative error							1		1				2
8.0 Regulatory restriction				2						1	1		4
9.0 External events									1			1	2
10.0 Environmental operating constraints Tech specs													
Total	18	13	13	12	11	13	7	7	7	8	8	7	124

general sequence of events here involved an instrument or piece of equipment that was out of service for repair. Once fixed, it was then returned to service. While it was being returned to service, another instrument (sensor, switch, etc) common to the sensing or instrument line was tripped and the reactor scrammed. The other three events (5-10-72, 5-23-72, 2-25-78) dealt with troubleshooting or investigative activities. In these troubleshooting activities, the personnel involved were searching for causes to particular problems and in the process inadvertently tripped the reactor.

Only two of the operational events were connected with procedural errors. The first incident which forced the unit down was an accidental boron injection on May 19, 1976. Before this the unit had been shut down for refueling. A surveillance test on the standby liquid control system (SBLC) explosive valve was conducted as the unit was preparing to start up. Because valving procedures were in error, 16% of the SBLC storage tank volume drained into the reactor vessel resulting in an accidential boron injection. This delayed the startup of the unit for five days. On April 27, 1978, a surveillance procedure did not make reference to a cross-tie between breakers in units 2 and 3. Opening this breaker caused an undervoltage condition on unit 2 resulting in a scram.

Eleven (or 1/3) of these operational events involved personnel performing incorrect action. All but two of these events occurred before 1975 while the other two occurred during 1975 and 1976.

Two of these events dealt with reactivity anomalies due to a faster cooldown rate than expected. The first (4-6-70) was due to the startup of an idle recirculation pump while the second (3-24-70) dealt with control rod insertion. Both of these events occurred before the first synchronizing of the generator to the grid in April 1970.

Three of these events dealt with valving errors. On October 19, 1973, a pneumatic air supply valve was closed which led to an inadvertent MSIV closure. On September 3, 1974, valving errors caused a generator-turbine mismatch. On September 29, 1975, a nitrogen bypass valve was left open resulting in high pressure in the containment.

Three of the events dealt with the transferring of control from automatic to manual or switching type of events. Two of these (5-28-70, 1-10-71) dealt with feedwater control, the other (11-3-76) with a recirculation pump. The general sequence of events was as follows: during the switching or transferring of control, a spike would occur in the flow instrumentation and cause the reactor to scram.

A major event in terms of outage time caused by incorrect operator action occurred on February 18, 1974 when the turbine generator turning gear was damaged because it was not properly engaged. Upon investigation, it was determined that the maintenance of a spring on the toggle mechanism had been faulty causing the turning gear to try to re-engage while the turbine was being rolled. This event kept the unit down for 329 hours.

Six of the events concerning operational error dealt with inadvertent actions leading to a reactor scram. On April 30, 1973 a premature switching from "startup" to "run" mode resulted in a scram. On July 9, 1977, during a routine surveillance test, a dead weight (undefined) was dropped on reactor level instrumentation causing a scram. On July 29, 1978 an operator using a walkie-talkie in the auxiliary electric room inadvertently caused the turbine 12% overspeed circuit to trip the reactor. On January 30, 1980, a half scram signal was received and the operator inadvertently scrammed the reactor. On May 22, 1980, an instrument rack was inadvertently jarred resulting in a reactor scram. Most of these occurrences at Dresden 2, unlike all the others, occurred during the latter years of its operating history.

4.4.4.2 Pipe cracks. Pipe cracks at Dresden 2 were a major problem and are discussed in detail in section 4.5.2.1. Most of the pipe cracks were in the coolant recirculation system. Likewise, almost 50% of the leaks not associated with the pipe cracks dealt with the seals of the recirculation pumps. This was a recurring problem throughout the operating history and not confined to a specific time period. There were only four instance of significant problems with condenser tube leakage which required the unit to shut down for repairs (3-30-70, 10-13-70, 6-12-72, 5-20-78).

4.4.4.3 Turbine control valves and turbine EHC oil leaks. Turbine control valves and the turbine EHC system experienced oil leaks quite frequently. The purpose for discussing this trend is because of the function of the EHC which is to slave the turbine to the reactor and thus assure a constant reactor pressure for a specific reactor power. These events appeared primarily in groups of occurrences about every three or four years beginning in 1972, 1975 to 1976, and 1980 (1972: 5-10, 6-22, 8-6, 12-14; 1975: 8-2, 9-10, 10-25, 11-15, 11-16, 11-22; 1976: 1-2, 8-21, 12-29; 1980: 10-9). Low oil pressure or oil leaks were the major problems although an EHC pump did malfunction in 1980.

4.4.4.4 Inadvertent closure of MSIV's. Inadvertent closure of MSIV's was a recurring problem. The general failure mode involved loss of compressed air to the pilot valve of the MSIV resulting in closure of the MSIV as it is designed to do. The significance of this problem is that these inadvertent closures resulting in a shutdown of the reactor represent unnecessary challenges to reactor safety systems and shutdown systems. Several things have been tried to correct this problem. The brass nipples on the fittings in the instrument air supply system were replaced with stainless steel nipples. The removal of the poppet valve and associated signal lines was considered to improve the reliability of the MSIV's. During this time period, many modifications to the instrument air system have been made including the replacement of several parts of the system with the drywell pneumatic air supply. The purpose of this was to obtain a better, cleaner, oil-free air supply.

4.4.4.5 Feedwater regulator valves. During a three year period from 1973-1976, six problems were encountered with feedwater regulator valves. Two the events (2-12-74, 2-18-76) dealt with either seal leaks or valve stems breaking. In one event (9-19-76) the regulator valve failed open due to a blown fuse with the control circuit. In the other event, the problems were unidentifiable (1-18-73, 2-19-73, 11-14-73). Low reactor water level resulted from these events (except in one case); however, a total loss of feedwater did not occur. This series of events is discussed due to the number of occurrences happening in a short period of time.

4.5 Reportable Events

The review of the operating history of Dresden 2 included a review of 625 reportable events. These were submitted to the AEC and NRC concerning technical specification violations and limited conditions of operation. The reports were filed as letters, abnormal occurrence reports (AOs),* reportable occurrences (ROs), and licensee event reports (LERs).

The reportable events filed for the years 1970 through 1981 have been compiled and categorized. This involved reviewing the report and recording the information in tabular form as per Sect. 1.3. These tables are arranged by year in Appendix A, Part 2.

Additional compilations were made from these tables during the analysis of the reportable events. Four areas of operating experience have been examined in detail. Efforts were augmented by additional sources of information such as docket material other than LERs, journal articles, and various NUREG reports on operating experience. The investigations undertaken are as follows:

1. The total number of reports filed by year was examined. This included evaluating possible trends and identifying non-system-specific recurring problems.
2. A system-specific review of the reportable events was made. This involved examining all events recorded by each system and examining events where failures were reported for specific systems.
3. A compilation of significant events was made and is discussed with event initiators and safety function failures highlighted.

4.5.1 Review of reportable events

From 1970 to 1981, 625 reportable events were reviewed for Dresden 2. A histogram of reportable events is illustrated in Fig. 4.1. As can be seen, a sharp increase of reports occurred in 1974. Prior to 1974 an average of 33 reports were filed per year while after 1973 the number had roughly doubled to 62. The following sections present a summary for each year of operating experience at Dresden 2.

4.5.1.1 Yearly summaries of reportable events.

1970

Dresden 2 was licensed for operation on December 22, 1969, and achieved initial criticality on January 7, 1970. Commercial operation did not begin until June 9, 1972.

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*The AO designation by Dresden 2 for identifying operational events during a particular time frame is not to be confused with those safety-significant events listed in the Report to Congress on Abnormal Occurrences (NUREG-0090 series).

Seven of the thirty events reported in 1970 involved the reactivity control system. The seven events involving the reactivity control system were: control rod drive (CRD) insertion times exceeded limits (four occurrences), sodium pentiborate crystals prevented a relief valve from closing (once), a CRD failed to insert (once), and a CRD failed to withdraw (once). The second most reported system during 1970 was the main steam isolation system (five events). The system failures were due to: an MSIV closing too slow (once), closing too fast (once), an MSIV pilot valve's temperature was too high (once), and four MSIVs failing to close (twice). The failure of the four MSIVs to close are significant events and are discussed in further detail in Sect. 4.5.2.2.

Another significant event occurred on June 5, 1970. A spurious turbine control system signal resulted in a turbine trip and reactor scram. Following the reactor scram, the safety valves inadvertently opened releasing primary coolant into containment. The plant remained shut down until August 11, 1970. For further details, see Sect. 4.5.2.3.

The startup power test program continued until December 7 when an internal failure in the main power transformer resulted in a plant shutdown. The unit 2 transformer was replaced with the unit 3 transformer and returned to service on December 7, 1970. This event is discussed in the forced shutdowns and power reduction sections (Sect. 4.4.1 and 4.4.2.2.2).

1971

The number of reportable events decreased to 29 in 1971. The high pressure coolant injection system (HPCI) dominated the system failures and was declared inoperable on three of the five occurrences. The first event occurred during a quarterly surveillance test on January 19, 1971. Results of the test indicated that HPCI pump flow was less than that specified in the technical specifications. A similar test performed on December 31, 1970 had the same results. However, the HPCI system was not declared inoperable as required. The calibration of a square root converter in the flow indication circuit shifted. This caused the indicated flow to be 450 gpm less than the actual HPCI pump flow.

The second failure in the HPCI system involved a radioactive release from the site when a pipe ruptured during a surveillance test. The test, conducted on May 27, involved pumping water from the condensate storage tank (CST) and returning it to the CST via a test line. Following initiation of the system, an unexpected decrease in the storage tank's water level was noted. HPCI was shut down. The water level dropped when the test return line to the CST ruptured upstream of a manual isolation valve. The manual isolation valve was left in the closed position. The aluminum test line was not designed for HPCI pump discharge pressures. A revision to the HPCI system valve checklist was made to include a check of the test line manual isolation valve's position.

In addition to the pipe rupture, the HPCI steam supply valve opened properly, but failed to close. Failure was due to a bent valve stem. During a surveillance test on August 5, the HPCI turbine stop valve failed to open. Failure of the stop valve to open rendered the system incapable of providing coolant injection as required by the technical specifications. A misaligned limit switch on the HPCI control valve, which is in the solenoid circuit for the stop valve, caused the valve to fail in the closed position. Realignment of the limit switch corrected the problem.

During a surveillance test on November 16, the HPCI system was declared inoperable. The hydraulic system would not reset from the control room, but could be reset manually. A loose limit switch on the control valve created an open circuit to the solenoid valve. This would have prevented an automatic start of the HPCI turbine. Corrective action consisted of tightening the adjusting screw and adding a double lock nut to prevent further loosening.

The plant's first refueling outage started in late February. Major efforts during the outage were directed toward locating and replacing 215 defective fuel assemblies. Examination via fuel sipping revealed 41 leakers of the remaining fuel assemblies. One was not sipped, one was disassembled, 25 had unverified fuel rod characteristics, and 147 were labeled "bad rod lot content." The labeling of fuel assemblies under the category "bad rod lot content" consisted of non-leakers identified for replacement based upon their potential for containing a population of internally contaminated fuel rods. The potential for failing was based upon the multiple early life failures of fuel rods in the same production lot. The selection of assemblies discharged was based upon removing as many fuel rods from bad lots as possible up to the limit of the number of new assemblies available for replacement (total of 215 assemblies).

The unit was returned to service on May 29, 1971 and power was increased in a stepwise fashion to 100% power for continued testing of selected plant systems. Within several weeks, off-gas activity measurements indicated additional fuel degradation. The plant was operated at a reduced power level for the remainder of the year.

No events considered significant occurred during 1971.

1972

The number of reportable events occurring at Dresden 2 increased to 31 events in 1972. On June 9, the plant began commercial operation.

The containment isolation system and reactor trip system accounted for eight of the thirty-four events (four each). Three of the containment isolation system violations were due to excessive leak rates while the fourth was a return sample isolation valve failing closed. Three of the reactor trip system failures resulted from set point drifts on pressure switches. The fourth event occurred when a turbine stop valve closed. A reactor trip was not initiated after the closure.

Two noteworthy events occurred during the year. On July 26, 1972, an operator failed to respond to an alarm when a sample pump in the plant chimney monitoring system tripped. The stack gas sample flow annunciator was in the alarm condition for seven and one-half hours. The importance of this alarm was emphasized with all persons involved.

The second event involved a partial flow blockage in one of the four main steam lines. The plant was shut down on September 7 to investigate. A flow restrictor was missing from the line and had lodged at the inlet of the inboard MSIV. No damage to the MSIV occurred. Modifications to the flow elements prevented future occurrences.

1973

The number of reportable events continued to increase as 43 events were reported in 1973. Eight of the reportable events occurred in the liquid waste management system. All eight of the events were reported when activity limits were exceeded in various radwaste and drain tanks.

Only one event occurred in the gaseous waste management system and was the first of four of this type of event to occur at Dresden. On March 27, an explosion occurred in the off-gas system, while preparations were under way to modify the system (Ltr., April 5, 1973). The 30 min holdup line, while being purged with air, leaked. A welder's torch ignited the hydrogen. The explosion flashed through the off-gas system and blew out the permanent filters on one end and the temporary filters on the other end. Low level particulate contamination resulted in the areas of the two filters. Two men that were injured by the explosion suffered no radioactive contamination. Consequently, all subsequent off-gas line purging proceeded after the area was roped off.

Another event worth noting involved logic errors in the circuit designs of five safety systems: standby liquid control system, standby gas treatment system, diesel generator auxiliaries, and two engineered-safeguard buses (Ltr. 3/7/73). The design of the circuits in each case was such that when one component was racked out, another component was prevented from operating. This same circuit design was utilized on both Dresden reactors and both Quad Cities reactors.

1974

The number of reportable events nearly doubled from 1973 to 1974 (43 to 70). Equipment failures in the reactivity control system, coolant recirculation system, and the feedwater system resulted in the second highest downtime in Dresden's operating history (see Sect. 4.4).

The most time consuming problem in the reactivity control sytem was the frequent uncoupling of control rods (four times). Additionally, a significant event occurred in this system on November 2. After the closure of a control valve, operators failed to receive an alarm or a half scram signal. Further details are given in Sect. 4.5.2.6.

Other time consuming equipment failures were pipe cracks which were discovered in the core spray system, containment isolation system, coolant recirculation system, and feedweater system. Due to the significance of pipes cracking in these systems, more detailed analyses can be found in Sects. 4.5.2.1 and 4.5.3.1.

For the second straight year, an explosion occurred in the off-gas system (February 20). The filter in the system had been previously grounded in order to preclude such explosions. Further investigation revealed that the filter core had two pieces of underground metal instead of one, and thus the initial modification was inadequate.

1975

Twelve of the fifty-five reportable events that occurred in 1975 involved the emergency generator system. Diesel generator failures accounted for 11 of these with 4 of the failures due specifically to failure to start.

No significant events occurred during the year, but there were two events worth noting. During the 1975 refueling outage, a withdrawal of control rods commenced while personnel were still in view of the core (AO 75-22). Personnel began control rod drive friction testing and were 75% complete before noticing the error. The master refueling procedures specified that no one must be within line-of-sight of the core during a control rod withdrawal. The friction test procedures did not explicitly state this. The procedures were modified to reflect this requirement.

For the third consecutive year, the off-gas system had an explosion (AO 75-35). While attempting to apply the sparging air, the operator discovered that the sparging air supply header was full of water. As the header was being drained, the preheater inlet pressure stood at 15 psig. The operator opened the recombiner outlet as specified by procedures. The pressure decreased and the operator returned control of the pressure to the sparging air flow controller. Three minutes later, the explosion occurred.

No specific cause of the explosion could be determined. A check of the recombiner system showed that the preheater inlet and outlet manual drains on both trains were closed. These valves should have been open. This blockage probably caused the 15 psig in the piping. All other valve positions were normal and it appeared unlikely that the recombiner system caused the explosion.

1976

The number of reportable events increased to 66 events in 1976. The reactivity control system accounted for 13 of the events while the gaseous radioactive waste management system accounted for 6 events.

Two significant events occurred during the year. During inspections of the jet pumps on March 27, loose restrainer clamp bolt keepers were found on 19 of the 20 jet pumps (RO 76-19). Each pump has two keepers and a total of thirty keepers were loose.

The second event occurred on May 25, 1976 (RO 76-34). During automatic blowdown surveillance testing, a relief valve remained open causing a continuous blowdown condition. When it became evident that the valve could not be closed, a manual scram was initiated. Further details of these events are provided in Sect. 4.5.2.

Four events of interest occurred during the year. The first event was a repeat of an occurrence in 1975. On April 13, personnel were not evacuated from within the line-of-sight of the reactor vessel (RO 76-25). The personnel affected were not directly involved in the fuel handling procedures. Consequently, they were not instructed to evacuate the area. In 1975, the control rod exercising program procedures were modified to prevent the occurrence of this type of event.

On May 25, 700 gallons of sodium pentaborate solution leaked into the reactor vessel (RO 76-31). During testing of the explosive injection valve and injection plug, the pump inlet valve from the boron tank was valved in. This was in accordance with procedures. The boron solution drained from the storage tank, through the pump and into the reactor vessel. The vessel water was discharged to the radwaste system.

The last two events involved the gas treatment system. A design review, completed on September 2, revealed that the swing SBGT could be disabled upon an auto-initiation signal in either of the other two units (RO 76-59). When one of the units produced an auto-initiation signal, the inlet valve to the other unit would close. This valve remained closed when the initiation signal was removed and reset. An auto-initiation signal from the second unit would in turn isolate the first unit, thus isolating both units. This circuit design was utilized on both Dresden reactors.

The last event involved a possible hydrogen explosion in the off-gas system making it the fourth hydrogen explosion in four years (RO 76-40). Personnel heard what they thought to be an explosion and initiated an immediate load reduction. Both rupture disks failed but the off-gas filters were found to be acceptable. The cause of the overpressurization was not determined but it was hypothesized that it resulted from a hydrogen explosion.

1977

The second largest number of reportable events occurred in 1977 (71 events). Failures in the reactor trip system (12 events) and in the emergency generator system (11 events) dominated the system failures. Ten of the twelve events in the reactor trip system were due to instrument drifts while five of the emergency generator system failures were due to failure of a diesel to start.

Two significant events occurred during the year. On July 9, 46 control rods failed to fully insert (RO 77-27). Of the 46 CRDs, RO 77-27 states that 22 CRDs failed in a similar manner on two previous occasions.

On December 2, emergency power was lost (RO 77-70). The unit 2/3 swing diesel was out of service and the unit 2 diesel would not start. The second diesel finally started after three attempts. Further details of these events are given in Sect. 4.5.2.

A chemical spill on August 5 degraded the control room atmosphere and resulted in a Generating Station Emergency Plan "on-site alert." Even though this event is not a reportable event, it is discussed due to its potential significance. The spill occurred when a plant operator was simultaneously filling the make-up demineralizer acid and caustic day tanks. His attention was diverted resulting in the overflow of both tanks. The chemicals mixed producing a high concentration of chemical vapors that prevented the operator from reaching the controls. The fan in the make-up room was not operating and the fumes eventually seeped into the control room ventilation system. The vent fans for the control room were turned off to minimize the inflow of fumes.

Several alterations to the plant resulted from this event. First, the exhaust fan for the make-up demineralizer room was repaired. Second, all penetrations in the ceiling and walls were sealed. Third, the HVAC ducts which pass through the demineralizer room were taped. The long-term corrective actions included installing a switch in the control room that would permit operation of the HVAC in either the recirculation mode or outside air supply mode. Additionally, the demineralizer day tanks were moved and high level trips were installed on all day tanks.

1978

Fifty-two reportable events occurred during 1978. Problems with the emergency generator system continued (8 occurrences). As in 1977, the main cause of system failure was failure of the diesel generator to start (5 occurrences). One of the significant events also involved the diesel generators (RO 78-50). The unit 2 diesel failed to start while the unit 2/3 swing diesel was out of service (see Sect. 4.5.2 for details).

Pipe cracks, which appeared to be a major problem in 1974, were still being observed (RO 78-46). On August 8, two cracked socket welds were found on a recirculation pump drain line. The cracks were caused by improperly fitted pipe joints. The pipe crack phenomenon is discussed in further detail in Sects. 4.5.2 and 4.5.3.

The third significant event to occur during the year when the isolation condenser was rendered inoperable while the HPCI system was out of service for repair. A switching error caused power to the bus which feeds the isolation condenser to be lost. Further details are given in Sect. 4.5.2.10.

1979

The number of reportable events increased to 59 during 1979. For the fifth consecutive year, the emergency generator system experienced a large number of failures (12 events). As in the previous years, failure of the emergency diesel generator to start was the most predominant mode of failure. The one event that was categorized as significant for 1979 was due to the failure of the diesels to start (RO 79-34). On May 24, the unit 2 diesel failed to start during an operability test subsequent to finding an LPCI valve inoperable. On May 30, the unit 2/3 swing diesel failed to start after the unit 2 diesel was declared inoperable. The operators initiated an immediate shutdown of the reactor. For further details regarding failures of the diesel generators, see Sect. 4.5.2.12.

An event of interest occurred on February 2 when a section of blowout panel at the west side of the refueling floor was blown out (RO 79-10). Secondary containment was breached. The airborne activity on the refueling floor was low. Therefore, a small amount of radioactivity could have been released to the atmosphere prior to restoring a negative pressure. The panel blew out after all of the exhaust fans tripped while the four supply fans continued to run. This pressurized the reactor building causing the blowout panels to fail.

1980

The number of reportable events in 1980 (45) was the lowest number of events since 1973. The recurring failures in the emergency generator system appeared to be solved as only two events occurred. Both of the failures were due to the diesel's heat exchanger leaking.

The reactivity control system experienced more failures than any other system (7 events). Four events were reported when a control rod uncoupled which had not occurred since 1977. The other three events in this system involved the CRD scram discharge volume. On June 11, it was determined that the correct seismic supports were not installed. Consequently, the CRD scram discharge piping did not meet seismic requirements. The second event involved the ultrasonic testing of the scram discharge volume not being performed on time. Lastly, on December 2 during a reactor scram, the CRD volume Hi-Hi level alarm failed to annunciate. This event was categorized as significant and is reported in more detail in Sect. 4.5.2.

1981

As shown by the histogram in Fig. 4.1, the largest number of reportable events occurred in 1981 (74 events). For the second year in a row, the emergency generator system experienced fewer failures than in previous years.

On December 23, the fire protection deluge system actuated in the HPCI room (LER 81-79). The HPCI system was declared inoperable. The event by itself was not categorized as significant since the ADS was still available. However, later the same day, an electromatic relief valve in the ADS system failed to open (LER 81-78). These events coupled together are considered to be significant and are discussed in detail in Sect. 4.5.2.

4.5.1.2 System summary of reportable events. A compilation of all reportable events by system and year is presented in Table 4.6. Systems having a large number of reportable events have been categorized at the subsystem level. The system codes CJ and SFA (see Table 1.2) have been assigned to the isolation condenser and the automatic depressurization system, respectively, since no codes are specifically provided for those systems from the list of codes used in NUREG 0161.[1] The code for core reflooding system was used for the automatic depressurization system (ADS) since ADS is a part of the emergency core cooling system, and the core reflooding function is provided by the low-pressure coolant injection system.

An examination of the data presented in Table 4.6 reveals no discernible time-dependent trends other than what is expected from random fluctuations in the data. Approximately 78% of the reports involved the following systems: ECCS (18.8%), reactor coolant (15.9%), containment (12.3%), emergency power (11.3%), radioactive waste management (10.5%), and reactivity control (9.6%). The emergency power and the reactivity control systems are unique subsystems with a sufficient number of reportable occurrences such that they were considered separately. The other systems are general system categories (see Table 1.2).

4.5.1.2.1 Emergency core cooling systems. The reportable events for the emergency core cooling system(s) were examined in order to obtain estimates of the systems' unavailability. The system was chosen for three reasons: first, the likelihood of the system to be challenged during an operational transient (loss of feedwater, loss of offsite power, stuck open relief valve etc.); second, the status of the system is usually discernible from the reports; and third, other systems unavailability estimates are readily obtainable for comparison. A system is defined to be unavailable upon demand if it fails to start or it fails to function satisfactorily. Assumptions and calculations of the failure rate are found

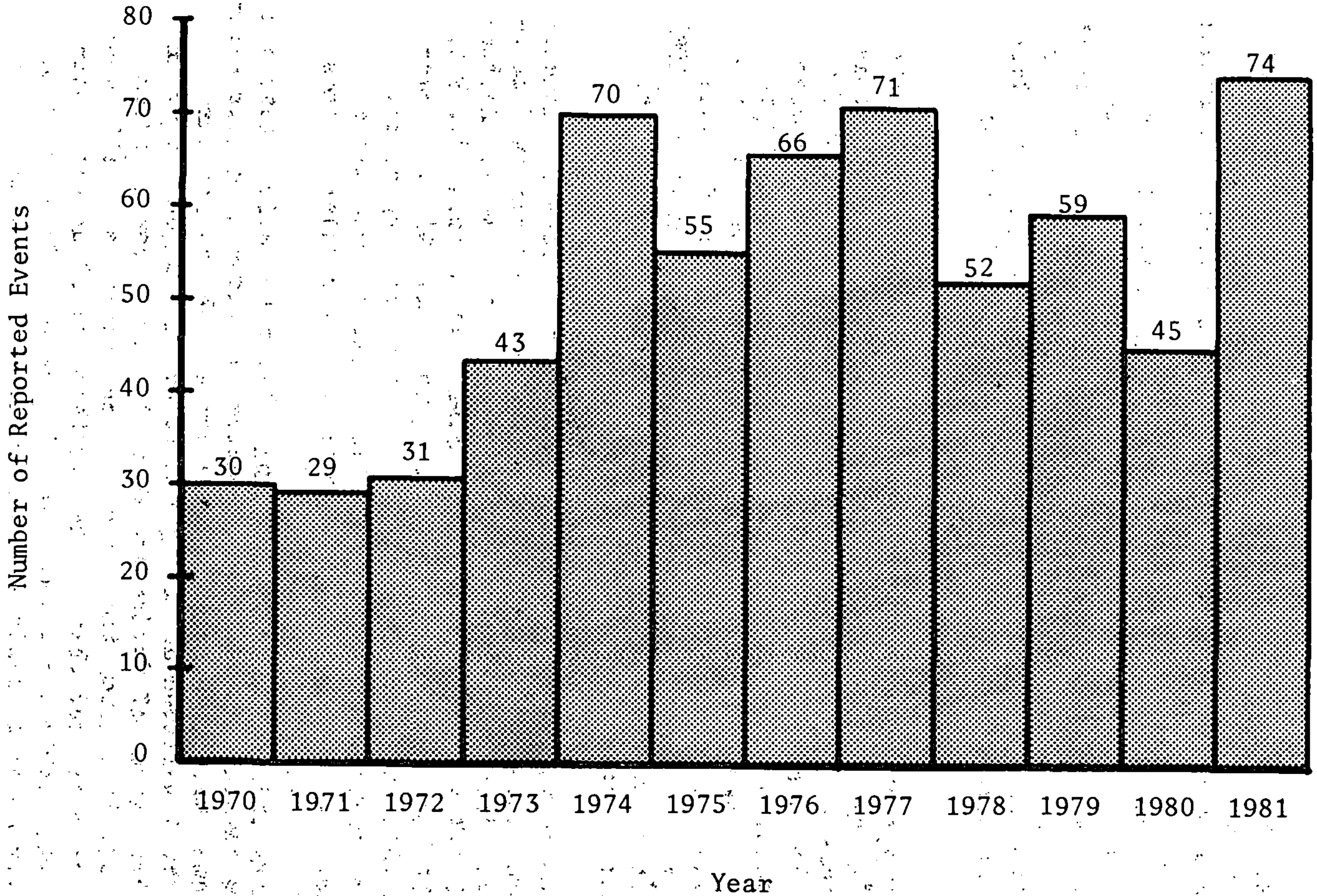

Fig. 4.1. Number of reported events at Dresden 2.

Table 4.6 Summary of systems involved in reportable events by year at Dresden 2

System	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	Total
Reactivity control (RB)	7	1	3	2	8	4	13	8	2	1	7	7	63
Reactor (RX)	1	3					1	2					9
Coolant recirculation (CB)		1			3			4	2	1		4	15
Main steam isolation (CD)	5	1	2	2	1	4		2	4	1	2	2	26
Residual heat removal (CF)		2	1	3	2		1	3			1		13
Feedwater (CH)				1	5		1	2		1		1	11
Other coolant subsystems (CJ)				3		1	2		1	2	2	2	13
Reactor coolant (CX)	3		1	1	8		1	1	1	3	2	5	26
Containment (SA)		2	3	1	2	4	1	8	1		3	6	31
Containment heat removal (SB)				1	1	2		1		1			6
Containment air cleanup (SC)		1							1				2
Containment isolation (SD)		1	4	1	7		3		7	6	2	4	35
Containment combustible control (SE)						2				2	2		6
Emergency core cooling (SF)		2			1	5	4	1			2	1	16
Core reflooding system (SF-A)	3				1				1			1	6
Low pressure safety injection (SF-B)		1		5	3	1	2	5	2	4	3	1	27
High pressure safety injection (SF-C)	3	5	2	7	6	5	7	5	3	2	3	4	52
Core spray (SF-D)		2			5	6	2		1	2		4	22
Other engineered safety features (SH)				2	6	1	2	1	1		1	1	15
Instrumentation and controls - general (IX)		2					4	1	4	1		2	14
Reactor trip (IA)			4	1	1	1	2	12	5	5	3	9	43
Engineered safety feature instrument (IB)			2	2	2	2	2	2		1	3	2	16
Electric power (EX)	3	3	1		1	1	1		1			3	14
Emergency generator systems (EE)	1	1	3	2	2	12	5	11	8	12	2	1	60
Fuel storage and handling (FX)									1	1			2
Auxiliary water (WX)				2		1	2	1					6
Auxiliary process (PX)	4					1	1					1	7
Other auxiliary systems (AX)										4	5	2	11
Steam and power (HX)			3				2		1	2	1		9
Liquid rad waste (MA)	3	4	1	8	5			3	2	5	1	4	38
Gaseous rad waste (MB)		2	3	1	4	2	6	1	2	1		4	26
Process and effluent monitoring (MC)												3	3
Radiation protection (BX)							1		3	2	1	1	8
System code not applicable							2						2
	33	34	33	45	74	56	69	74	54	60	46	75	653

in Appendix B. It must be emphasized that no attempt was made to bound the Dresden 2 estimates; thus a conclusion cannot be drawn from the following comparison alone.

The emergency core cooling system consists of the following systems: ECCS, core reflood (or automatic depressurization), LPCI, HPCI, and core spray. Approximately 18.8% of all reportable events involved one of these systems (or 123 events). Concern, however, was placed on the HPCI system as there were 52 reportable events filed concerning this system. Approximately 29% of these were failures upon demand (15 events). The remaining reports involved instrument drift, maintenance activity, and piping problems. About 48% of the Dresden HPCI failures involved motor operated valves, turbine stop valves, or isolation valves. Problems with instrumentation and controls accounted for an additional 15%. The HPCI failure rate showed no discernible time trend.

Estimates of the failure rate of the HPCI system for Dresden 2 indicate a failure rate several times that predicted in the Reactor Safety Study[15] and a factor of two greater than that observed from historical data.[16]

Only two ADS failures were observed (AO 75-41 and LER 81-78). Both involved failure of two ADS valves. Historical data indicate valve failures to be the major contributor to ADS unavailability.[18] Sufficient data, however, was unavailable to make a meaningful comparison beyond this. This is also felt to be true for LPCI and core spray systems.

4.5.1.2.2 Reactor coolant system and connected systems. The reactor coolant system and connected systems accounted for 15.9% of the reports filed (104 events). A large fraction of these reports involved set point drift, miscalibration of instrumentation, and inadequate response times of equipment. In other events, redundant valves and breakers failed to open or close, and tripped on demand. These items are of minor safety significance.

Reactor coolant system problems of greater significance generally involved the main steam isolation system. Main steam isolation pilot valves failed on several occasions resulting in MSIVs failing to close. Details of these events are given in Sect. 4.5.2.2.

Two events of particular interest involved the main steam isolation system and the isolation condenser. Dresden 2 was shut down on August 7, 1972, when it experienced a partial flow blockage in one of the four main steam lines (Ltr. 9/7/72). Operators discovered that the downstream cone of the flow restrictor was missing from the line and had lodged at the inlet of the inboard MSIV.[17] The MSIV was not damaged. Fatigue at the locations of poor weld penetrations caused the failure. Visual examination of the same cone-to-throat section weld on the flow elements for the other three lines over the inside and outside surfaces showed no indications of surface cracking. However, these flow elements were modified with lateral supports to prevent vibration, and with axial pins to prevent the cone from being swept downstream on failure of the weld. The plant returned to service at a reduced power level with both isolation valves closed on the line that experienced trouble. The reduced level continued until a replacement flow element was installed.

The second event was considered significant and involved the isolation condenser. On March 21, 1978, power to the bus feeding the unit 2 isolation condenser was inadvertently switched off (RO 78-24). At the time, the high pressure coolant injection system was out of service for repair. Further details of this event are given in Sect. 4.5.2.

4.5.1.2.3 <u>Containment systems.</u> The containment systems accounted for 12.3% of the reports filed (80 events). Nearly one-half of the reports involved valve failures (38 events). The majority of the valve failures exhibited excessive leak rates (13 events). Instrument drift and miscalibrating set points also contributed to system failures (7 events).

The majority of containment related problems were attributed to inherent failure (60%). An additional 28% resulted from operational error, i.e., administrative, maintenance, or operator error. Another 6% involved design errors including the torus coating problem.

Problems with the torus coating were discovered during the 1971 refueling outage. An inspection revealed some pinpoint rusting, blistering, and delamination of the coating above the water line. Upon draining the torus, personnel discovered extensive degradation of the torus coating below the water line. Large areas of delamination occurred in which the coating had separated from the primer coat and a thin film of water formed between the two coats. The coating was eventually removed and replaced with an inorganic zinc coating.[18]

4.5.1.2.4 <u>Emergency power system.</u> Diesel generator failure was the dominant contributor to degraded and failed states of the emergency power systems. Forty-four of the seventy-four events occurring in the emergency power system were failures upon demand. Twenty-two of these failures were failures to start upon demand. A best estimate of the failure rate upon demand appears in Appendix B. The failure rate for Dresden 2 was roughly twice that reported in the Reactor Safety Study.[15] However, the estimate for Dresden 2 did fall within the upper and lower bounds.

Roughly one-third of the failures were associated with the air starting system. An investigation of problems with the air starting system during 1979 resulted in a modification to the air starting circuitry. The modifications allowed multiple starts to be attempted prior to locking out the starting sequence. The previous design allowed only one start attempt before the starting sequence was locked out. No failures to start were reported in 1981 with only one failure to start reported in 1980. This is opposed to seven reported failures to start in 1979 and five in 1978.

No loss of offsite power events were identified in this review. The only cause of an emergency power system failure was the loss of both diesel generators. On three occasions both the unit 2 and the unit 2/3 swing diesel failed to start or were inoperable simultaneously. However, one incident (RO 78-50) was not considered an emergency power failure since the diesel was immediately reset and started successfully.

On December 2, 1977, Dresden 2 was completing its refueling outage with the diesel generator cooling water pump out of service (the unit 2/3 swing diesel failed on 11/29). An operability surveillance test on the unit 2 diesel generator failed (RO 77-70). Both air start motors engaged and the diesel turned over until the low pressure on the air receiver terminated the starting sequence. Apparently, the engine did not receive any fuel. A second start attempt also failed. Adjustments were made to the fuel system prior to a third attempt, which was successful. The cause of the problem, however, was never determined. As a preventative measure, the test frequency was increased from monthly to weekly. This event was identified as a precursor to a significant accident.[16]

The second event occurred on June 12, 1979 (RO 79-34). The unit 2/3 swing diesel generator failed to start during an operability test after the unit 2 diesel generator was declared inoperable. The lower Bendix air starter failed to engage. Since the emergency power system was inoperable, a plant shutdown was initiated. This event was also identified as a precursor to a significant accident.[16]

4.5.1.2.5 Radioactive waste management system. The radioactive waste management system accounted for 10.3% of the reports filed (67 events). The reports generally involved the liquid and gaseous radioactive waste management systems. The majority of reports involving the liquid radioactive waste system concerned either radiation limits being exceeded in storage tanks (17) or small unplanned releases (13). Administrative and/or operator errors caused most of these events. Problems in the gaseous radioactive waste management system involved mechanical component failures. Explosions in the off-gas system were of special concern in the early- to mid-1970s. The problem was resolved by modifying the environment the undiluted offgas hydrogen-oxygen mixture was exposed to.[19,20] This included:

1. replacing the stellited stainless steel with bronze in selected components,
2. increasing the closure time of the recombiner flow control valve to reduce the spark potential,
3. properly grounding valve internals and filters,
4. removing the condenser recirculating line heater from service, and
5. reducing the pressure on the air ejectors.

Details of the hydrogen explosions are given in Sect. 4.5.3.

4.5.1.2.6 Reactivity control system. Problems with the reactivity control system accounted for 9.6% of the reportable events (63 events). The majority of the events involved the uncoupling of one or two control rods. Slow control rod insertion time also provided aggravation during the plant's early life. This resulted in a design modification to the control rod drive inner filter. Two other problems of potential concern involved the failure of several control rods to fully insert and the failure of the scram discharge volume hi-level alarm.

On July 9, 1977, 46 control rod drives failed to fully insert (RO 77-27). All of the control rods latched at position 02; however, if all the rods were in the same bank, this event could be symptomatic of the scram discharge volume problem experienced at Browns Ferry in 1980.[21] The failure was attributed to worn or damaged stop piston seals. No other gross failures of control rods to fully insert were reported. For further details, see Sect. 4.5.2.8.

The second incident occurred on December 29, 1980. During a scram, the scram discharge volume hi-hi level alarm was not received. The cause of the failure was attributed to: excessive cable lengths and routing, less than optimal transducer placement and mounting arrangement, redundant transducer signal interference. The system was tested satisfactorily after alterations were made to eliminate the formal cause.

4.5.1.3 Cause summary of reportable events. The causes of reportable events are tabulated by year in Table 4.7. On several occasions two causes were assigned to the same event. The percentage of reportable events by cause is graphically depicted in Fig. 4.2. Inherent failures dominated the causes of reportable events with administrative, design, maintenance, and operator errors accounting for 10 to 20% each. Human errors accounted for roughly half of all reports with the contribution of in-plant and out-of-plant personnel-related errors being approximately equal. In-plant personnel errors involved hands-on human involvement such as installation, maintenance, or operator errors and in most instances directly involved the reactor operating staff itself. Out-of-plant personnel errors involved administrative, design, and fabrication errors. These events were generally attributed to the reactor or component vendor, the A/E, or utility management.

4.5.1.4 Events of environmental importance. A summary of radioactivity releases for Dresden 2 and 3 is shown in Table 4.8. The table lists the airborne and liquid releases and the solid waste shipped for the years 1970 through 1981. The activity for the solid waste shipments are for Dresden 1, 2, and 3 combined. Dresden reports releases and solid wastes as a station versus unit by unit.

Forty-six events involved radiation limits being exceeded, radioactivity releases, or personnel exposure. Seventeen events concerned activity levels in storage containers exceeding allowable limits (Table 4.9). The other 26 events involved radioactivity releases or personnel exposure (Table 4.10). Eleven events concerned gaseous or liquid releases beyond the plant boundary. Eight of the releases were due to human errors. Human errors represent the releases that resulted from administrative errors, operator errors, maintenance errors, or combinations thereof. The other events involved potential or real exposures to plant personnel. If procedures had been followed or were more clearly defined, the majority of exposures could have been avoided. Four exposures require further elaboration.

During the 1975 refueling outage (March 21), operators began control rod drive friction testing while other personnel were within "core line-of-sight." The master refueling procedurers specified that no one is allowed to be within the core line-of-sight during a control rod withdrawal. The operator completed 75% of the test before realizing the error. Procedures were modified to expound upon this requirement. On March 13, 1976, control rods were withdrawn again with personnel within the core line-of-sight. The personnel affected were not directly involved in the fuel handling procedures. Consequently, they were not instructed to evacuate the area.

The other two events concern definite exposures to plant personnel. On March 5, 1981, a licensee contractor employee received a whole body exposure of 21 rems while guiding a crane in the removal of temporary concrete shielding from inside the reactor vessel. The water level beneath the shielding had apparently dropped and was not detected by a control room monitoring device. Radioactive components exposed by the low water

Table 4.7. Causes of reported events by year at Dresden 2

Cause	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981	Total
Administrative error	5	4	2	13	7	5	7	9	5	3	5	2	67
Design error	2	4	4	8	12	8	6	6	3	3	2	2	60
Fabrication error	2	2	2		7		2	3		1			19
Inherent failure	16	15	18	14	32	25	39	39	30	35	29	51	343
Installation error			1	2	6	5	6	6	2	5	2	3	38
Lightning													0
Maintenance error	4	2	2	7	5	8	9	9	10	6	4	5	71
Operator error	4	3	2	1	2	7	3	1	5	6	3	11	48
Weather	1												1
Total	34	30	31	45	71	58	72	73	55	59	45	74	647

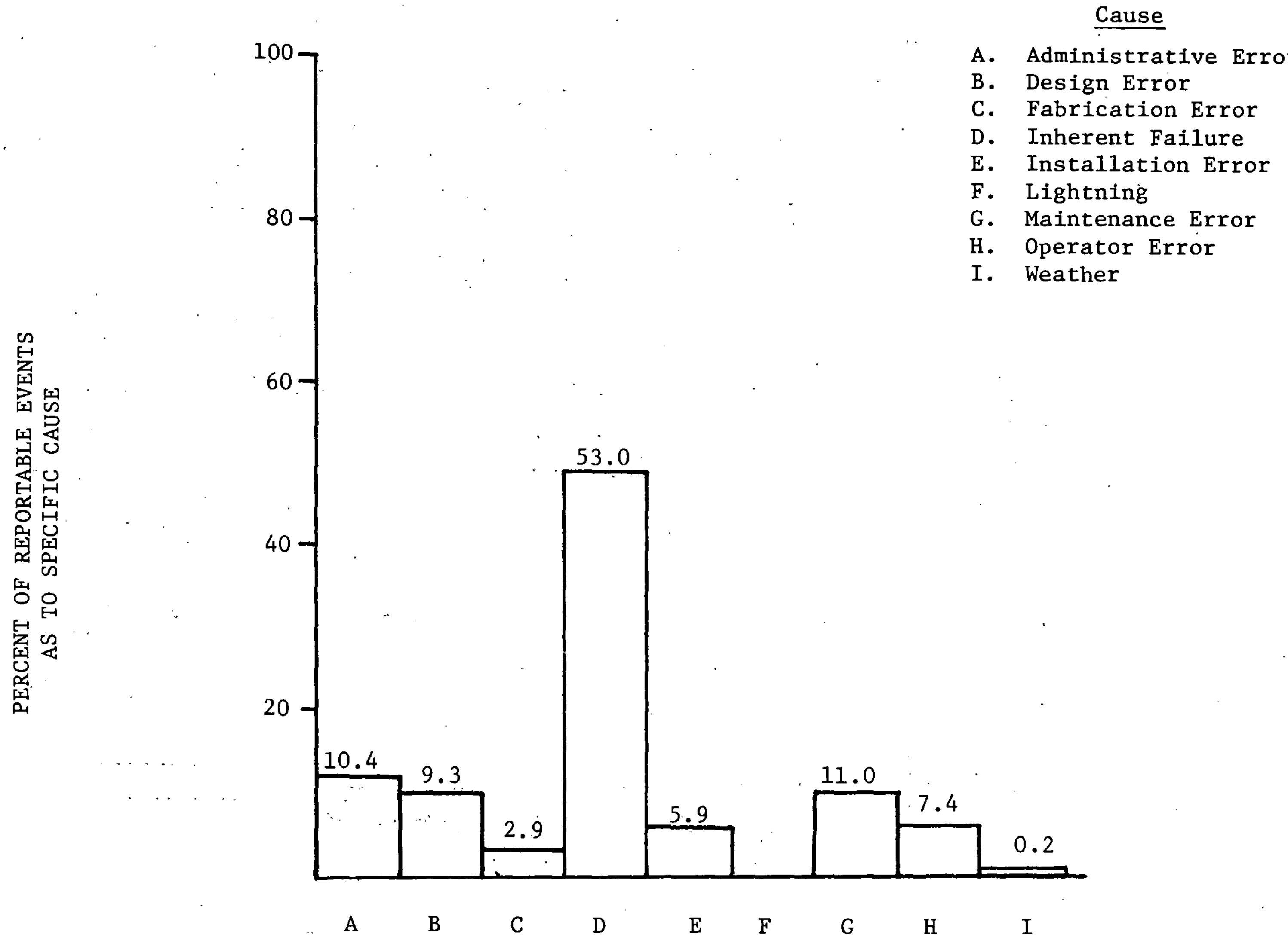
PERCENT OF REPORTABLE EVENTS
AS TO SPECIFIC CAUSE
100
80
60
40
20
10.4
9.3
2.9
53.0
5.9
11.0
7.4
0.2
A
B
C
D
E
F
G
H
I
Cause
A. Administrative Error
B. Design Error
C. Fabrication Error
D. Inherent Failure
E. Installation Error
F. Lightning
G. Maintenance Error
H. Operator Error
I. Weather

Table 4.8 Summary of radioactivity released from Dresden 2 and 3[a]

	1970	1971	1972	1973	1974	1975	1976	1977	1978	1979	1980	1981[c]
EFFLUENT (CURIES)												
Airborne:												
Total noble gases		5.80 E+5	4.29 E+5	8.80 E+5	6.27 E+5	3.69 E+5	3.23 E+4	3.13 E+5	4.06 E+4	6.91 E+4	3.65 E+4	2.25 E+4
Total I-131	NA	NA	NA	4.90 E+0	3.86 E+0	8.13 E-1	1.43 E+0	2.33 E+0	7.09 E-1	1.40 E+0	9.68 E+1	1.80 E+0
Total halogens (including I-131)	NA	NA	NA	2.45 E+1	3.80 E+1	1.17 E+1	1.28 E+1	5.22 E+1	5.67 E+1	3.46 E+1	1.26 E+2	2.66 E+1
Total particulates ($T_{1/2}$ > 8 d)	1.60 E+0	8.68 E+0	5.89 E+0	2.50 E+0	2.83 E+0	3.50 E+0	4.51 E+0	4.53 E+0	2.42 E+0	5.57 E+0	7.48 E+0	5.22 E+0
Total tritium	NA	NA	NA	1.00 E+1	1.14 E+2	2.21 E+2	1.66 E+2	5.00 E+2	3.25 E+2	1.85 E+2	1.18 E+3	2.00 E+2
Liquid:												
Total mixed fission and activation products		2.30 E+1	2.20 E+1	1.59 E+1	3.31 E+1	8.10 E-1	1.21 E+1	4.40 E-1	3.99 E-1	2.65 E-1	3.65 E-1	2.92 E-2
Total tritium	3.10 E+1	3.85 E+1	2.59 E+1	2.58 E+1	2.26 E+1	5.40 E+1	1.97 E+1	5.00 E+0	1.92 E+1	1.94 E+1	3.08 E+1	5.27 E+0
Dissolved noble gases	NA	NA	NA		1.44 E-1	3.10 E-2	None	None	None	None	None	None
Solid:												
Total	NA	NA	NA	1.34 E+2[b]	5.05 E+3[b]	7.74 E+3[b]	4.33 E+3[b]	1.13 E+4[b]	1.88 E+3[b]	8.45 E+2[b]	5.62 E+3[b]	1.98 E+3[b]

[a]Releases are reported for Dresden as a station, and not unit specific.

[b]Includes Dresden 1, 2, and 3.

[c]January 1, 1981–June 30, 1981.

Table 4.9. Activity levels in storage containers exceeding allowable limits of Dresden 2

Number	NSIC accession	Event date	Cause	Description
	60226	9/27/70	A	Waste sample tank exceeds activity limit of 0.7Ci (1.8 Ci)
	71709	5/17/72	H	High level of radioactivity in waste tanks
	80116	3/20/73	A	Radwaste surge tank exceeds activity limit of 0.7 Ci (5.6 Ci)
	80117	3/21/73	A	Activity in radwaste tank exceeds 0.7 Ci
	80496	4/2/73	A	Activity in radwaste tank exceeds 0.7 Ci (1.05 Ci)
	80113	4/5/73	A	Activity in radwaste tank exceeds 0.7 Ci
	-	4/9/73	A	Activity in radwaste tank exceeds 0.7 Ci
	82276	7/9/73	A	Activity in floor drain tank exceeds 0.7 Ci (1.04 Ci)
	82277	-	A	Activity in floor drain tank exceeds 0.7 Ci (0.8 Ci)
	83163	8/2/73	A	Activity in floor drain tank exceeds 0.7 Ci (0.89 Ci)
	91671	5/24/74	A	Activity in radwaste surge tank exceeds 0.7 Ci
	92441	5/28/74	A	Activity in above ground tanks exceeds 0.7 Ci
AO 74-49	96270	9/24/74	A,B	Activity in above ground tanks exceeds 0.7 Ci (0.94 Ci)
RO 78-27	138923	4/18/78	A,H	Activity in storage tank exceeds 0.7 Ci
LER 79-43	151174	6/21/79	H	Activity in floor drain sample tank exceeds 0.7 Ci (0.74 Ci)
LER 79-66	153495	12/13/79	A	Activity in floor drain sample tank exceeds 0.7 Ci (0.708 Ci)
LER 80-06	154927	1/19/80	-	Activity in waste sample tank exceeds 0.7 Ci (1.5 Ci, 0.9 Ci)

Table 4.10. Events involving radiation releases or personnel exposures at Dresden 2

Number	NSIC accession	Event date	Cause	Description
	57053	10/15/70	A,H	The radwaste concentrator had leaks, RO blew down
	56981	10/15/70	A,H	The boiler water was released to the river (0.327 μCi/L)
	65542	3/2/71	A	Release rates out the stack exceeded limits
	63266	–	D	Fuel pin leakers, off-gas limits not exceeded
	63267	5/27/71	H	HPCI test line ruptured, release to surrounding area
	66505	8/7/71	H	Improper valving arrangement resulted in release
	64655	8/8/71	H	Improper valving arrangement resulted in release
	72822	7/26/72	D	Stack gas sample pump tripped while operating (not a test)
	80133	3/27/73	A	Explosion in off-gas system. 1.76 Ci of Xe-133 released out stack
	91118	4/27/73	D	Acid induced leak in radwaste piping
	94181	6/20/74	H	Radioactive release to discharge canal exceeds limits
	94799	7/22/74	B,A	LPCI valve failed, release to surrounding area
74-70	97739	11/9/74	D	Air ejector ruptures in rad gas system
75-27	103078	4/21/75	H	CR withdrawn with personnel within core line of sight
76-25	114373	4/13/76	A	CR withdrawn with personnel within core line of sight
76-38	114644	5/26/769	G,H	Chimney sample valve left open twice
76-40	115732	6/7/760	D	Explosion in off-gas system, 2.57 Ci out chimney
76-60	118429	9/5/76	E	HPCI test line leaks due to bad weld
76-61	-	9/15/76	G	HPCI test return line laks, water leaked to sewer
77-12	126203	3/21/77	B	Radwaste floor drain surge tank sump overflowed
77-20	125592	5/29/77	D	Heat exchanger tube leaked resulting in liquid release
77-56	131761	11/4/77	D	Heat exchanger tube leaked resulting in release to pond

Table 4.10. continued

Number	NSIC accession	Event date	Cause	Description
79-10	147356	2/2/79	B	Refueling panel blow-out panel was blown out
79-40	150361	6/5/79	D	Radwaste reboiler leaked, contaminated the soil
-	-	3/5/81	A	A contractor received 21 rem while guiding a crane
-	-	1/1/81-3/20/81	A	A contractor received an exposure over the quarterly 3 rem (3.02)

level set off alarms and the area was evacuated. Although there were other workers in the area at the time of the incident, only the one worker received an overexposure.[22]

The second exposure involved a contractor employee that received a cumulative radiation exposure of 3.02 rems for the time period January 1 through March 20, 1981.[22,23] This exceeded the NRC limit for radiation workers. The limit is 3 rems per calendar quarter.

Overall, human errors contributed significantly to the radiation releases and personnel exposures (19 events). Administrative errors and operator errors dominated the causes of said events (14 events). Design errors, maintenance errors, and installation errors also contributed under the category of events caused by human errors.

4.5.2 Review of significant events

Each reportable event was screened as a step in the evaluation processes (see Sect. 3.2). A compilation of the significant events by year at Dresden 2 is given in Table 4.11. Events with serious safety implicatons are described in detail in the following sections. Table 4.12 presents a brief description of these events. Those events which degraded a safety function or initiated a DBE are: pipe cracks (4.5.2.1), main steam isolation valve failures (4.5.2.2), release of reactor coolant into containment (4.5.2.3), RPS signal failed (4.5.2.4), safety relief valve failed (4.5.2.5), 46 control rods failed to fully inset (4.5.2.6), cracks in jet pump restraints (4.5.2.7), isolation condenser rendered inoperable (4.5.2.8), ADS valve failed while HPCI was inoperable (4.5.2.9), emergency power system failures (4.5.2.10), and the scram discharge volume high level alarm failed (4.5.2.11).

4.5.2.1 Pipe cracks. The occurrence of pipe cracks did not appear to be of safety significance until 1974. During 1974, cracks were discovered in the core spray system (AO 74-73), containment isolation system (AO 74-75), coolant recirculation system (AO 74-76, AO 74-77), and the feedwater system (AO 74-35). The core spray system had three cracks identified in the "A" core spray loop and two in the "B" loop. The cracks ranged from 1/8 to 3/4 in. in length. Fourteen bellows in the primary containment system also exhibited cracking. The coolant recirculation system appeared to be the most susceptible system to pipe cracks. A crack on the 4-in. "B" recirculation bypass loop was on the upstream side of the equalizing valve. This made repairs relatively easy since the line could be isolated. The "A" recirculation loop required freeze plugs since the line could not be isolated through the use of valves. These cracks were discovered in September 1974. Additional cracks were discovered during a reexamination of the recirculation loops in December. Finally, a 4-in. crack was identified on a 10-in. feedwater line. Additional cracking in the recirculation loops were identified in 1978 (RO 78-46). The pipe cracking phenomenon observed at Dresden 2 is evaluated further in Sect. 4.5.3.1.

4.5.2.2 Main steam isolation valve failures. Two significant events involved the main steam isolation system. Main steam isolation pilot valves failed on several occasions resulting in MSIV failure to close. On

Table 4.11 Tabulation of events categorized as conditionally significant C4, C5, C7, and C8

NSIC Accession Number	Report number	Description	Section discussed in
<u>C-4 - Design, manufacturing or fabrication deficiency</u>			
	Ltr 3/8/71	LPCI valve fails to auto-open under loss of offsite power conditions	
79601	Ltr 3/7/73	Common cause problem in the Engineer Safety System logic	4.5.4
81478	Ltr 460-73	Reset spring missing in breaker	
98584	Ltr 12/24/74	LPCI recirculation loop break detection requires additional additional dampening	
98585	Ltr 12/19/74	Incorrect type of switch utilized in MSL high temperature sensor	
112725	RO-76-16	Through wall crack in HPCI pipe safe end	4.5.3
113286	RO-76-21	Through wall cracks in isolation condenser safe end	4.5.3
118185	RO-76-59	Logic error renders SBGT plant inoperable	4.5.6
131788	RO-77-61	Reactor water level switch drift	
137855	RO-78-28	Modification to prevent spurious closure of recirculation loop valve	
<u>C-5 - Reports involving long outages or major equipment damage</u>			
93663	AO-75-11	Through wall cracks in core spray injection line	4.5.3
100046	AO-75-12	Through wall cracks in core spray piping	4.5.3
100047	AO-75-13	Through wall crack in core spray piping	4.5.3
113284	RO-76-19	Cracks in jet pump restraints	4.5.5
115064	RO-76-31	Sodium pentaborate leaks into reactor vessel	4.5.6

Table 4.11 (Cont.)

NSIC Accession Number	Report number	Description	Section discussed in
<u>C-8 - Other reports considered conditionally significant</u>			
63141	Ltr 4/29/71	Six seismic snubbers found broken due to water hammer	4.5.6
63200	Ltr 3/21/71	Torus point scaling	4.5.2.5
63267	Ltr 6/4/71	HPCI test return line ruptures	4.5.6
68334	Ltr 11/1/71	Water hammer jars piping	
93512	Ltr 1/12/75	Cracks found in feedwater spargers	4.5.3
93602	AO-75-4	D.G. fails to start after 6-h run due to improper setting of drop switch	
94749	AO-75-26	Interlocks prevent 3 LPCI valves from opening during test	4.5.6
95427		Pipe cracks on recirculation lines	4.5.3
96465		Safety valve actuates prematurely	4.5.6
96534		One-inch pipe leaks at bad weld joint	4.5.6
103078	AO-75-27	Control rod withdrawn with personnel in core line-of-sight	4.5.6
103085	AO-75-44	Role of tape dropped in the reactor annulus	--
133691	RO-77-72-36	D.G. fails to close onto breaker	4.5.2.5.1
138256	RO-78-30	Incorrect valving renders LPCI/CS degraded	4.5.6
140065	RO-78-42	Turbine control valve fails to fast close	4.5.5
140512	RO-78-46	Two cracks in recirculation pump suction line	4.5.3
140514	RO-78-44	APRM set out of limits	4.5.6

Table 4.11 (Cont.)

NSIC Accession Number	Report number	Description	Section discussed in
140582	RO-78-35	Cracks found in spent fuel storage tanks	4.5.3
149352	LER-79-23	Bad weld in reactor head vent line	4.5.6
149728	LER-79-45	Cracks in feedwater piping and welds	4.5.3.3

Table 4.12. Tabulation of reports categorized as significant at Dresden 2

NSIC Accession Number	Report number	Description
S2 - Two or more failures due to a common cause occur during the same event		
47289	Ltr 5/18/70	The 1C, 2C, 2A, and 2D MSIVs failed to closue during testing
60727	Ltr 12/11/70	Four MSIVs fail to close
S4 - Component failures occur that would have easily escaped detection		
113284	RO 76-19	Cracks in jet pump restraints
S5 - An event proceeded in a way significantly different from what would be expected		
47814	Ltr 7/6/70	Major blowdown event
S7 - An event that could have been a greater threat to plant safety		
137335	RO 78-24	Isolation condenser inadvertently rendered inoperable
172107	LER 81-078	ADS valve failed while HPCI was inoperable
S9 - Other events considered significant		
94914	AO 74-35	Four inch crack in 10-in. line
95593	AO 74-46	75% crack in recirculation line
97078	AO 74-59	Failure of RPS signal
98517	AO 74-77	Additional pipe cracks on recirculation line
98577	AO 74-75	14 primary containment bellows leak
98579	AO 74-73	Pipe cracks found in core spray
114645	RO 76-34	Safety relief valve fails to close
127977	RO 77-27	46 control rods fail to insert
-	RO 77-70	Emergency power was lost
140153	RO 78-46	Two cracks found in recirculation pump welds
140162	RO 78-50	Emergency power was lost
150051	LER 79-34	Emergency power was lost
163371	LER 80-46	Scram discharge volume high level alarm fails

May 8, 1970, the 1D, 2A, 2B, and 2C MSIVs failed to close and the 1A and 2D valves failed the timing test (Ltr. 5/18/70). The problem was traced to fouling of the pilot valves. It is believed the fouling was caused by particulates in the air supply to the pilot valves. The air supply was blown down as a corrective action. On December 4, 1970, four MSIVs again failed to close due to fouled pilot valves. On January 22, 1971, one MSIV failed due to an oil fouled air pilot valve.

Industry wide, main steam isolation valve failures have been primarily related to the following causes: poor quality control air to the pilot valves and binding of the MSIV valve stems with the valve stem packing. These two failure modes contributed to about 85% of the MSIV failure to close events industry wide. Both causes also represent common-mode failure mechanisms.

These two failure modes are significant in that they identify mechanisms by which more than one MSIV may fail to close at the same time thus leading to conditions which have not been considered in the plant's safety analyses, and they are continuing to occur even though corrective actions reported in the LERs indicate that technology is available to prevent such failures.[24]

Dresden 2 has experienced the fouling of the pilot valves due to particulates in the air supply. However, it appears that the corrective action resolved the problem as it was limited to the plant's early life.

4.5.2.3 Release of reactor coolant into containment. In all instances except one, the engineered safety features functioned properly to bring the reactor to a safe shutdown. The one event where engineered safety features failed to function properly occurred on June 5, 1970. A series of multiple failures complicated by operator error and procedural inadequacies contributed to the significance of the event.[25,26] With the reactor undergoing initial startup tests and operating at 75% power (623 MWe), a spurious signal generated in the electrohydraulic control of the turbine-generator set caused the turbine control valve to open further and the steam bypass valves to the condenser to fully open. Within one second the turbine tripped and the reactor scrammed. The two operating feedwater pumps tripped due to low suction pressure due to the increased feedwater flow. Subsequently, the MSIVs closed and the water level control in the pressure vessel became difficult. Water level began rising again, but because the level-indicator chart pen being observed by the operator stuck, the operator further increased the flow rate of feedwater not knowing the level was still increasing. By the time the operator discovered the stuck pen, the water level had risen enough to flood the main steam lines and the isolation condenser steam line. The incident was further complicated at this point by a lack of procedural guidance under conditions of high reactor coolant in the pressure vessel. The continued input of water coupled with afterheat from the reactor core and closure of the main-steam-line valves caused the pressure-vessel pressure to begin increasing rapidly. The isolation condenser system was actuated manually, but it was shut off immediately due to a too-low trip setting of the condensate-return-line flow required by an erroneous technical specification. An attempt to reopen the main-steam-line valves to dump steam through the turbine-bypass valves failed because the valves had not been reset from the earlier trip that had closed them. Following the automatic tripping of the recirculation pumps and automatic startup of the standby diesel generators, the low-pressure spray- and coolant-injection systems started but did not inject water because the reactor pressure exceeded the pump head of both systems. The high-pressure coolant-injection system started but did not inject water, because it had been valved out earlier for repairs after proof-testing its backup systems as provided for in the technical specifications. Actual water injection by this system would have been automatically inhibited by the high-water signal from the pressure-vessel water-level monitors. With the isolation condenser inoperable, the operator manually opened a pressure-relief valve several times throughout the incident to dump steam to the pressure-suppression pool to reduce the pressure in the pressure vessel in order to remove the reactor decay heat. The high-pressure coolant released from this valve impinged on the lifting levers of two other safety valves and partially opened them. They remained open until they were closed manually after the vessel was depressurized and cooled down. Several thousand gallons of primary water leaked to the drywell. The containment zone was contaminated, but no measurable radioactivity was released to the site or the environs. Damage to the plant was minor.

4.5.2.4 Reactor protection system signal failed. During a weekly turbine test on November 2, 1974, neither an alarm nor half scram was received when a control valve was closed (Ltr. 11/12/74). Parameters indicated that the valve had closed. It was discovered that a lead had broken on the plug from the fast acting solenoid. The pins in the plug were too small for the wire being used. All plugs were replaced.

4.5.2.5 Safety relief valve failed. A safety relief valve failed to close during ADS testing resulting in a continuous blowdown of the reactor on May 25, 1976 (RO 76-34). The reactor was manually scrammed when it was determined that the valve would not reset. The failure was caused by excessive leakage on the pilot stage of the valve. Both pilot and secondary stages of the valve were replaced. As a preventive measure, the pilot stage of the valve was to be leak tested during every refueling outage.

4.5.2.6 Forty-six control rods failed to fully insert. Following a reactor scram on July 9, 1977, 46 control rod drives failed to fully insert and latch (RO 77-27). The CRD manual control system was utilized to fully insert the CRDs. Of the 46 CRDs, 22 failed in a similar manner previously. As on the previous occasions, worn or damaged piston seals were thought to be the cause of the problem. The worn piston seals allowed excessive reactor water leakage through to the buffer hole area of the CRD. As the drive piston closed the buffer during upward drive movement, the excess water could not be vented through the buffer hole fast enough. When the scram valves closed, the drives settled and latched at the "02" position. The safety significance of the event was minimized since analyses indicated that the reactor could be shut down safely in a hot standby condition with all control rods inserted at position "02" and the strongest rod completely withdrawn.

4.5.2.7 Cracks in jet pump restraints. An event that could have gone unnoticed involved structural supports for the jet pumps (RO 76-19). This event was discovered during a refueling outage. On April 16, 1976, it was reported that 30 loose restraining clamp bolt keepers were found on 19 of the 20 jet pumps. Each pump has two clamps. The failures were attributed to vibrational fatigue eroding. During the following refueling outage, another jet pump restrainer was found with an eroded weld (RO 77-42). Again, the cause was attributed to vibrational fatigue. As a corrective action, the restraint was relocated and rewelded.

4.5.2.8 Isolation condenser rendered inoperable. On March 21, 1978, the isolation condenser system was inadvertently rendered inoperable (RO 78-24). While unit 3 was undergoing a test, power to the bus feeding the unit 2 isolation condenser was inadvertently switched off. Since the high pressure coolant injection system was also out of service for repair, the plant's high pressure safety function was lost. The significance of this event was minimized by the shortness of its duration; however, this event has been identified as a precursor to a more serious accident.[16]

4.5.2.9 ADS valve failed while HPCI was inoperable. On December 23, 1981, a "diesel fire pump running" alarm annunciated in the control room (LER 81-79). Investigation revealed that the fire protection deluge system had actuated in the HPCI room. The HPCI system was declared inoperable and the required surveillances conducted. The cause of the fire system actuation was a high concentration of humidity and dust particles which actuated the detector.

This event by itself was not a significant event even though it posed a potential concern for reactor safety. With the HPCI system declared inoperable, the ADS was still available for high pressure operation. However, later the same day, an ADS electromatic relief valve failed to open (LER 81-78). A wire became wedged in the contact that bypassed the hold coil, causing it to be placed in series with the pickup coil. This prevented solenoid operation. The concern for reactor safety resulted from HPCI or ADS being required for a loss of normal auxiliary power or a small line break plus a loss of normal auxiliary power with a standby diesel available.[27]

4.5.2.10 Emergency power system failures. Loss of both diesels was the only cause of emergency power system failures. On three occasions both the unit 2 and the unit 2/3 swing diesel failed to start or were inoperable simultaneously. One of the incidents (RO 78-50) was not considered an emergency power failure however, since the diesel was immediately reset and started successfully.

On December 2, 1977, the unit 2/3 swing diesel generator cooling water pump was out of service while Dresden 2 was being refueled (the unit 2/3 swing diesel had failed on 11/29). An operability surveillance test on the unit 2 diesel generator failed. Both air start motors engaged and the diesel turned over until the low pressure on the air receiver terminated the starting sequence. Apparently the engine did not receive any fuel. A second start attempt also failed. Adjustments were made to the fuel system prior to a third attempt, which was successful. The cause of the

the problem, however, was never determined. The test frequency was increased from monthly to weekly as a preventative measure. This event was identified as a precursor to a significant accident.[16]

The final significant event involving emergency power system failures occurred on June 12, 1979 (RO 79-34). The unit 2/3 swing diesel generator failed to start during an operability test after the unit 2 diesel generator was declared inoperable. The lower Bendix air starter gear failed to engage. After the second failure, the lower start motor was replaced. Since the emergency power system was inoperable, a plant shutdown was initiated. This event was also identified as a precursor to a significant accident.[16]

4.5.2.11 <u>Scram discharge volume high level alarm failed.</u> During a reactor scram on December 2, 1980, the scram discharge volume continuous water level monitoring system hi-hi level alarm was not received (LER 80-46). The cause of the failure was attributed to excessive cable lengths and routing, less than optimal transducer placement and mounting arrangement, and redundant transducer signal interference. The system was tested satisfactorily after alterations were made to eliminate the formal cause. This event was also related to the problem with the scram discharge volume level instrumentation.[28] After the December 2 scram, investigations determined that the ultrasonic detectors were inadequately coupled to the SDV piping. The coupling was improved and testing showed that the system could detect water flow into the SDV during single control rod scram tests. On December 4, the reactor again scrammed. The high level annunciator alarms were not received immediately following the scram. Instead, the alarms annunciated when the SDV was draining. A low signal-to-noise ratio caused the level detection system's operability problems. Arrangements were made for replacement of components (including transducers) to improve the signal characteristics and reduce system noise. The problems were not detected earlier since in situ testing of the system was not performed. Instead, system operability was tested by tripping each alarm channel while its sensor was connected to a calibration standard consisting of a section of 4-in. pipe identical to the SDV piping.

4.5.3 <u>Trends and safety implications of reportable events</u>

As an additional step in the overall evaluation process, the reportable events at Dresden 2 were examined to find discernible trends that indicate potential safety problems. The following specific trends and problem areas were identified: (1) pipe cracks, (2) diesel generator failures, (3) control rod drives, (4) off-gas system explosions, and (5) HPCI system failures.

4.5.3.1 <u>Pipe cracks.</u> Pipe cracks have been found to be a generic problem in BWRs. Pipe cracks have occurred in austenitic stainless-steel piping in BWR nuclear power plants since 1965. These cracks were infrequent and did not occur at an abnormal rate until after September 1974, when cracks were found in the recirculation loop bypass line of Dresden 2. This, and additional cracks at Dresden 2, led the AEC to request several additional inspections of BWR plants. No additional cracks were found in the first inspection, but subsequent inspections led to the discovery of numerous cracks.[29] The intensive scrutiny in 1974 and 1975 may account for the frequent pipe crack reports since previously existing cracks may have been found only under this vigorous inspection. The large number of pipe cracks led to the conclusion that stress-corrosion cracking is a generic problem with BWRs. This has not been found to be a problem with PWRs.

The cracks at Dresden 2 occurred mainly in the bypass and recirculation piping, and safends. A few were found in the control drive lines. Cracks have also been found at the reactor vessel in the core spray nozzle and the feedwater inlet. The nozzles are composed of thick low-carbon steel with a stainless steel casing. The cracks were removed with local grinding.

The factors that cause the cracking problem are thought to be stress, water chemistry, and material. Stress reduction is difficult to obtain and monitor and therefore not practical at this time. Experience has shown that dissolved oxygen in the coolant is the major contaminant, and it causes intergranular stress-corrosion of sensitized stainless steel. The steel is sensitized in the heat-affected zones of the welds. The sensitized area is then susceptible to intergranular corrosion since it is under stress and in contact with high oxygen content water.[30]

The Pipe Cracking Study Group (PCSG) was formed by the NRC to investigate this problem. The PCSG stated that the oxygen level could be reduced by addition of ammonia to the bulk coolant, but this technique has not yet been demonstrated to be economically feasible.[29] The most promising solution is to replace the susceptible piping with material that will be less adversely affected by highly oxygenated water. Materials suggested are ferritic steels, Inconel 600, stabilized stainless steel types 347 and 321, cast stainless steels types 304 and 316, and types 304L or 316L stainless steels.[31] Improvements in water quality and stress reduction, wherever possible, were also recommended in combination with better materials. It was concluded that the future use of regular grades of types 304 and 316 stainless steel in BWR piping should be avoided.[32]

Efforts to resolve this problem are continuing. The recent crack rate in all BWRs is unknown. Therefore the overall effectiveness of the corrective action cannot be known. In the case of Dresden 2, the number of cracks reported has declined since 1974 and 1975, when six cracks were reported each year, as shown in Fig. 4.3.

4.5.3.2 Diesel generator failures. Diesel generators were involved in 59 reportable events. Forty-four of the failures were failures upon demand with 22 of these being failure to start. The swing diesel accounted for 23 failures upon demand with 16 being failures to start. This is important since the swing diesel is common to units 2 and 3. Several losses of emergency power occurred to unit 3 when the swing diesel failed and the unit 3 diesel also failed. Three emergency power failures occurred at unit 2 when the swing diesel failed and the unit 2 diesel also failed. These are discussed in detail in Sect. 4.5.2.10.

From 1975 to 1979, the diesel generators failed almost nine times per year. This compares to an average of two failures per year for 1970 to 1974, 1980, and 1981. The increased frequency of failures appears to be a result of the increased test frequency (monthly to weekly on December 2,

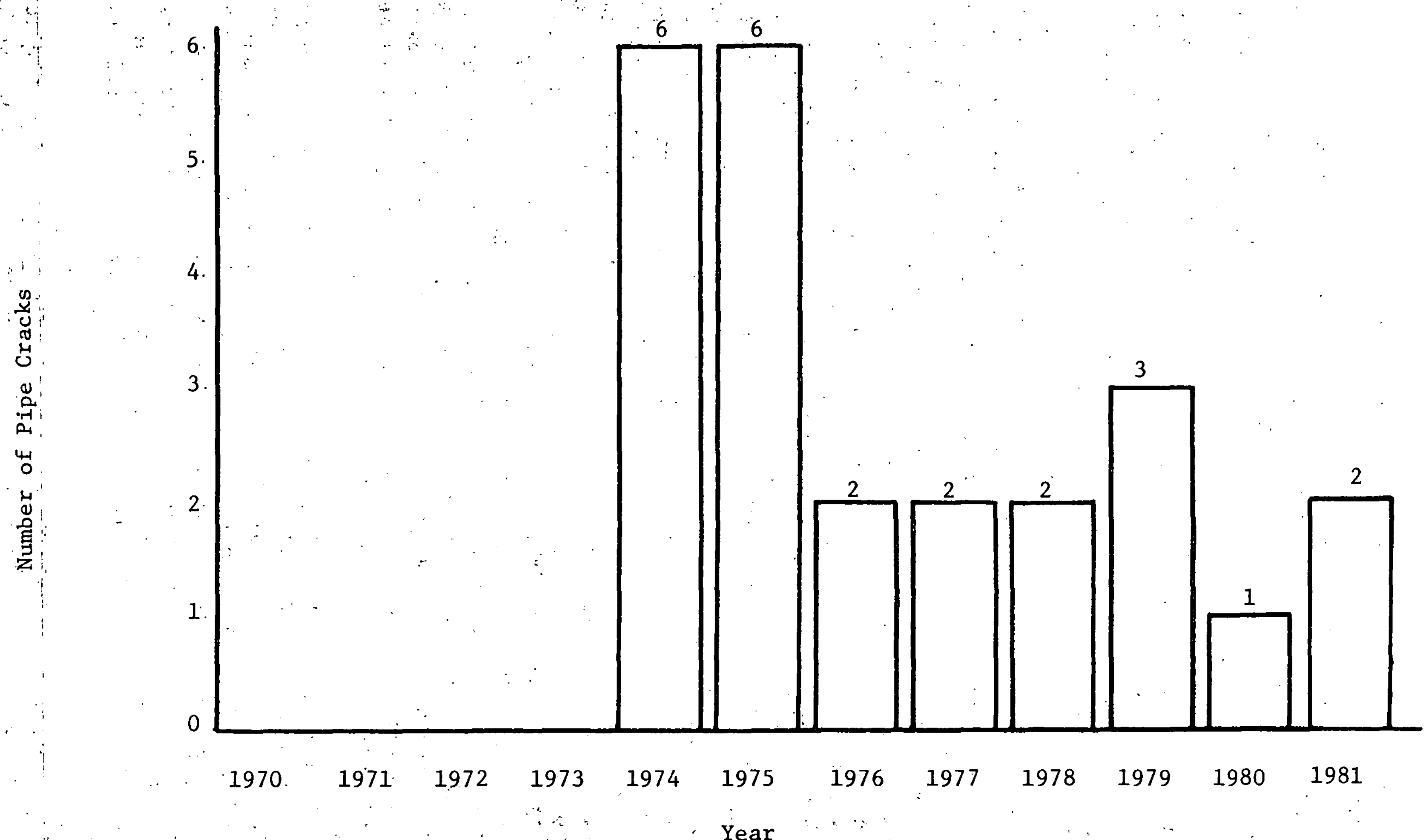

Fig. 4.3. Number of pipe cracks at Dresden 2.

1977). Estimation of unavailability was complicated by varying test frequencies. A best estimate of the failure rate upon demand appears in Appendix B. Based upon this estimate, the diesel generator failure rate upon demand at Dresden 2 is 0.057, or a factor of two greater than the median number reported in Wash-1400.[18] However, this estimate is still within the lower and upper bounds of the WASH-1400 estimate (0.01 to 0.1, respectively). Also estimated in Appendix B is the failure rate of the emergency power system. This is a factor of five greater than that reported in WASH-1400. The design analyzed in WASH-1400, however, utilized four diesel generators rather than three as at Dresden. This could influence the difference in numbers. The Dresden 2 estimates of the emergency power failure rate does agree with what is observed historically for all Boiling Water Reactors for the years 1969 through 1979.[16]

4.5.3.3 Control rod drives. A total of 40 reportable events occurred involving the control rods and the control rod drives. The peak years were 1970 (5 events), 1974 (5 events), 1977 (7 events), 1980 (8 events), and 1981 (6 events). Slow control rod insertion times provided aggravation during the plant's early life; however, a design modification to the control rod drive inner filter corrected this. Uncoupling of one or two control rods dominated the failures for 1974, 1977, and 1980. The failures in 1981 were dominated by excessive insert times.

4.5.3.4 Off-gas system explosions. Off-gas system explosions occurred at Dresden 2 for four consecutive years beginning in 1973. In 1973, the system leaked and the hydrogen was ignited by a welder's torch as he was cutting an overhead pipe. The explosion flashed through the off-gas system and blew out the permanent filters on one end and the temporary filters on the other. Low level particulate contamination resulted in the areas of the two filters.

The explosion in 1974 occurred due to inadequate grounding of the filter cores. The system had been previously grounded in order to preclude such explosions. Investigation revealed that the filter core had two pieces of underground metal instead of one.

No definitive cause could be determined for the 1975 explosion. Several valves that should have been open were closed. This probably caused the preheater's pressure to increase. However, it appeared unlikely that the recombiner system caused the explosion. Damage was limited to the rupture diaphragm and the off-gas filter.

In 1975, personnel heard what they thought to be an explosion. Both rupture disks were found failed but the off-gas filters were found to be acceptable. The cause of the overpressurization was not determined but it was hypothesized to result from a hydrogen explosion.

Explosions in the off-gas systems were inherent problems in boiling water reactors in the early- to mid-1970s. The problem at Dresden was resolved by modifying the environment that the undiluted off-gas hydrogen-oxygen mixture was exposed to.[19,20] This included:

1. replacing the stellited stainless steel with bronze in selected components,

2. increasing the closure time of the recombiner flow control valve to reduce the spark potential,
3. properly grounding valve internal and the filter,
4. removing the condenser recirculating line heater from service, and
5. reducing the pressure on the air ejectors.

4.5.3.5 HPCI system failures. Fifty-two of the 625 reportable events involved failures in the HPCI system (Table 4.13). Fifteen of these represented failures of the HPCI on demand. Forty-eight percent (twenty-five events) resulted from valve failures with the valves usually being motor operated valves, turbine stop valves, or isolation valves. Human errors caused more failures than inherent equipment failure (such as the previously mentioned valve failures). Overall, human errors contributed to 52% (27 events) of the HPCI failures. Maintenance errors (9), design errors (7), and installation errors (6) dominated the causes of the 31 human errors. There were more human errors than total number of events due to human errors as three events resulted from multiple errors.

Nine of the failures in the HPCI system involved the systems piping. Pipe cracks have been found to be a generic problem in BWRs. An AEC request for additional inspections of BWR piping in 1974 led to the discovery of many pipe cracks at Dresden 2. Prior to 1974, only three events were reported involving HPCI piping. None of these necessarily involved cracking; however, a test line did rupture in 1971. In 1974, a test line required repair and a bypass valve was discovered to be installed backwards. Three of the five events occurring after 1974 involved leaks in pipes or welds. Seven of the events were due to human errors with design errors dominating the cause (3 events).

The HPCI failures were evently distributed throughout the plant's operating history. Of the 15 failures upon demand, none occurred in 1981. Estimates of the HPCI system's failure rate indicated a failure rate several times greater than that predicted in the Reactor Safety Study. Additionally, the observed failure rate for Dresden 2 was twice the historical failure rate for all BWRs (see Appendix B).

4.6 Evaluation of Operating Experience

As discussed in Sect. 2, the main sources of information utilized during this review were forced shutdowns, power reductions and reportable events. The analysis included reviewing 206 forced shutdowns and power reductions as well as 625 reportable events.

Human operational errors were identified as the area of greatest significance in the review of forced shutdowns. These events include: incorrect action by plant personnel, events induced during maintenance and testing, and inadvertent actions. Roughly 20% of all shutdowns were caused by human operational error with about a third of these occuring during the first year of operation. Generally it appears that operational errors decreased with time until 1976. After 1976, they occurred at a constant rate. The exception to this is events induced inadvertently by personnel action, all of which occurred since 1975. About a third of these events occurred during surveillance testing. Incorrect personnel

Table 4.13. HPCI system failures at Dresden 2

NSIC accession number	Number	Event date	Cause	Description
–		4/70	G	HPCI pump supply valve inadvertently opened
46459		5/28/70	A,H	Water hammer due to improper valve arrangement
56054		9/70	B	Wrong size criteria in lubrication line
59490		1/19/71	D	Square root extraction malfunctioned
63267		5/27/71	H	HPCI test line ruptured
64047		8/5/71	D	HPCI turbine failed to reset
65545		8/5/71	D	HPCI turbine stop valve failed to open
67849		11/16/71	D	Loose adjusting screw on HPCI control valve
69791		4/72	B	HPCI flow switch found missing paddle and 2 screws
76405		11/6/72	A	HPCI MOV fails to open
80495		3/15/73	G	HPCI isolation valve fails to open
83226		8/15/73	D	Two HPCI MOVs fail
83229		8/15/73	D	HPCI steam supply valve fails to open
85319		10/17/73	G	HPCI valves loose seal
87047		11/23/73	B,E	HPCI DC power supply grounded
86995		11/27/73	A	HPCI test causes torus water level to exceed limit
87035		12/3/73	G	HPCI steam trip leaks
89392		2/23/74	D	HPCI drain line requires repair
95380	AO 74-40	8/15/74	C	HPCI pressure switch drift
97498	AO 74-63	11/11/74	B	Temperature probe fails
97718	AO 74-66	11/14/74	D	HPCI CST suction valves fail
97740	AO 74-69	11/20/74	D	HPCI MOV fails
95027	AO 74-37	8/1/74	G	HPCI failed test after maintenance
93514	AO 75-09	1/23/75	E	HPCI valve failed to open completely
100583	AO 75-15	3/7/75	D	Set point drift in HPCI high flow switch
–	AO 75-45	9/29/75	D	Cold solder joint causes HPCI failure to trip
106984	AO 75-48	10/7/75	E	HPCI fails to trip during transient

Table 4.13. continued

NSIC accession number	Number	Event date	Cause	Description
107982	AO 75-51	10/31/75	H	Operator error results in high torus water level
112166	RO 76-09	3/13/76	D	HPCI supply valve fails to open
112725	RO 76-16	3/24/76	B,C,E	Through wall cracks in HPCI pipe safe end
113285	RO 76-20	3/29/76	D	HPCI check valve leaks
115898	RO 76-43	6/21/76	E	HPCI level sensors damaged during construction
118429	RO 76-60	9/5/76	E	HPCI test line leaks due to bad weld
–	RO 76-61	9/15/76	G	Water from HPCI test return line leaked into sewer
–	RO 76-66	11/13/76	D	HPCI injection valve had severed stem
127976	RO 77-30	8/2/77	G	HPCI valve fails to open
129831	RO 77-35	9/10/77	D	HPCI valve fails to open
129832	RO 77-36	9/10/77	D	HPCI turbine's speed govenor fails
131796	RO 77-59	10/26/77	A	HPCI surveillance test missed
133617	RO 77-80	12/16/77	G	HPCI steam exhaust line leaks
137250	RO 78-23	3/20/78	B	HPCI control valve leaks
137830	RO 78-26	3/30/78	D	HPCI taken out of service for maintenance
141203	RO 78-55	9/30/78	D	HPCI trips on low suction pressure
149862	LER 79-02	1/3/79	D	HPCI fails to start
150360	LER 79-42	6/8/79	D	Set point drift in HPCI high steam flow switch
157698	LER 80-17	5/12/80	D	HPCI isolation valve fails to close
157686	LER 80-18	5/12/80	G	HPCI isolation valve fail's to close
160566	LER 80-39	10/11/80	D	HPCI steam supply valve fails to open
164684	LER 81-13	3/2/81	B	HPCI pipe supports require modifications
166566	LER 81-33	6/2/81	D	HPCI oil pump failed
168894	LER 81-57	9/3/81	D	HPCI inoperable due to broken hangers
172255	LER 81-79	12/23/81	D	Fire protection system actuation renders HPCI inoperable

action, inadvertent errors and maintenance activities caused the remaining shutdowns. Very few problems with incorrect procedures were identified.

Mechanical malfunctions most often cited as the cause of shutdown involved problems with: the turbine electro-hydraulic control system (EHC) the feedwater regulator valves, and the recirculation pump seals and bearings. Problems with the EHC system appeared to occur every three to four years (1972, 1975 and 1976, and 1980). This could be symptomatic of end of life failures for EHC oil seals. Feedwater regulator valve failures were clustered in a three-year period from the years 1973 to 1976. Failures of the recirculation pump seals and bearings appeared to occur at a constant rate.

Events involving in-plant personnel (as defined in Section 4.5.1.3) accounted for roughly 24% of all reportable events. No time-dependent trend was detected in the data. An additional 23% of the reports were caused by out-of-plant personnel (as defined 4.5.1.3). Again no time-dependent trend was discernible. Thus, human error consistently accounted for about one half of all reportable events. Mechanical problems of the greatest safety concern involved the HPCI system, the diesel generators, and general pipe cracking. Problems with HPCI and the diesel generators however appeared to be within the range of what one would expect to see based upon historical data. Diesel generator failures to start have decreased since the starting logic modification in 1979; however, two years (1980 and 1981) are insufficient to indicate that the problem has been solved. Pipe cracking in BWRs is a generic problem. Corrective action has been recommended by the pipe crack study group as discussed in section 4.5.3.1. This problem appears to be under control.

Dresden 2 experienced one major transient on June 5, 1970. This event is considered significant in terms of the SEP reviews and has been identified as a precursor to a more serious accident. Only one total loss of feedwater was identified in the review and no loss of off-site power events occurred.

In summary, six areas of operation should be of continued concern. These areas consist of two types: (1) those identified by either Dresden 2 or NRC and continue to recur — diesel generator failures, control rod and rod drive malfunctions, and radioactive waste management/health physics program problems, and (2) those areas of operation that have not been directly addressed by Dresden 2 or NRC — operator errors, turbine control valve and EHC problems, and HPCI failures. All six event types have continued to recur throughout Dresden 2's operating history.

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Appendix A: Dresden 2

Part 1. Forced Shutdown and Power Reduction Tables

Table A1.1 Forced Outages and Power Reductions for Dresden 2

No.	Date (1970)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	3/1	3	0		Spurious trip of IRM #12 while withdrawing control rods on an approach to critical.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N2.4
2)	3/1	2	0		Spurious trip of IRM #11.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N2.4
3)	3/2	5	0		Spurious trip of IRM #11.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N2.4
4)	3/2	82	0		Spurious trip of IRM #11.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N2.4
5)	3/24	2	0		Scram caused by IRM trips due to cold water addition when an idle recirculation pump was started.	G	3	Reactor Coolant (CB)	Pumps	D1.6
6)	3/30	137	3		Shutdown by insertion of control rods to locate and repair leaking tubes in the main condenser.	A	2	Steam & Power (HC)	Heat Exchangers	N3.1
7)	4/6	51	3		While inserting control rods, the cooldown rate was sufficient to add enough reactivity to cause scram due to IRM trips.	G	3	Reactor (RB)	Control Rods	D4.3
8)	4/11	16	2		Group 4 channel "B" scram solenoids failed to reset following a surveillance test. Test of channel "A" resulted in scram of group 4 control rods. Remainder of control rods inserted by manual scram.	A	2	Reactor (RB)	Control Rods	N1.14

Table A1.1 (Continued)

No.	Date (1970)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
9)	4/12	2	3		During heatup reactor pressure rose above 600 psi before 23" of vacuum established resulting in scram.	G	3	Instrumentation & Controls (IA)	Heat Exchangers	N6.1
10)	4/16	5	2		Test engineer accidentally caused reactor feed pump runout while installing test equipment. Sudden increase in relatively cool feedwater flow caused a positive reactivity addition and a scram from APRM high flux trips.	G	3	Reactor Coolant (CH)	Pumps	D1.2
11)	4/27	8	20		Test engineer accidentally introduced a signal into a pressure regulator causing a step decrease in setpoint. This resulted in the bypass valves opening and the resultant high steam flow caused an isolation and a reactor scram.	G	3	Reactor Coolant System (CC)	Valves	D1.7
12)	5/8	66	30	LTR: 5/18/70	A turbine trip from high water level combined with low vacuum caused a reactor scram. While subcritical MSIV test conducted with 4 valves failing. Orderly cold shutdown initiated.	A	3	Steam & Power (HC)	Heat Exchangers	D2.5
13)	5/14	5	54		Both operating reactor feed pumps tripped on low suction pressure caused by high condensate demineralizer differential pressure. Reactor scrammed to minimize the level transient.	A	3	Reactor Coolant (CH)	Pumps	D2.7

Table A1.1 (Continued)

No.	Date (1970)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
14)	5/19	4	75		During mainstream isolation valve testing at 75% power one MSIV closed causing high flows in the remaining three lines which resulted in a Group 1 isolation and reactor scram from MSIV closure.	A	3	Reactor Coolant (CD)	Valves	D2.4
15)	5/21	4	52		Test engineer accidentally caused an erroneous level input to the feed-water control system. The control system raised level to the trip point and with load above 45% turbine stop valve closure caused a reactor scram.	G	3	Reactor Coolant (CH)	Instrumentation & Controls	D1.2
16)	5/21	1	0		A spurious trip from IRM #14 caused a scram while on approach to critical.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N2.4
17)	5/23	5	52		A sensor failure caused turbine to trip, with load above 45% turbine stop valve closure caused the reactor to scram.	A	3	Steam and Power (HA)	Instrumentation & Controls	D2.3
18)	5/26	5	54		Loss of essential service MG set caused a loss of the feedwater control system. Reactor water level decreased to scram point and caused a reactor scram.	A	3	Reactor Coolant (CB)	Generators	D2.7
19)	5/28	4	60	LTR 6/2/70	While transferring feedwater control from auto to manual a level increase tripped the turbine. Reactor scrammed from stop valve closure . During the time of the scram, the HPCI steam line was out of service for repair and inspection. During the scram the water level rose and filled HPCI steam line. When HPIC put back into service the	G	3	Reactor Coolant (CH)		D1.2

Table A1.1 (Continued)

No.	Date (1970)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
19)	(Continued)				next day, hydraulic shock (water hammer) occurred causing damage to piping restraints and support.					
20)	6/1	6	50		Test engineer accidentally caused a turbine trip while installing test equipment. Closure of stop valves caused a reactor scram.	G	3	Steam & Power (HA)	Turbines	D2.3
21)	6/3	4	0		Reactor brought subcritical by insertion of control rods to allow drywell entry for SRM work.	B	1	Instrumentation & Controls (IA)	Instrumentation & Controls	N1.1.4
22)	6/5	1417	75	LTRs: 6/12/70 7/6/70 6/15/70	A spurious turbine control system signal resulted in bypass valves opening and a turbine trip. Reactor scrammed from turbine stop valve closure.	A	3	Steam & Power (HA)	Turbines	D1.7
23)	8/4	3	0		Spurious trip of IRM #16.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N2.4
24)	8/7	39	0		Scram caused by IRM #11, 12 &16 tripping on Hi-Hi.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N2.0
25)	8/28	7	75		Turbine tripped due to high water level caused by feedwater control valve stuck open. Reactor scram on stop valve closure.	A	3	Reactor Coolant (CH)	Valves	D1.2

Table A1.1 (Continued)

No.	Date (1970)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
26)	9/11	6	64		Instrument mechanic working on the controlling vessel level sensor introduced a faulty signal to the level controller. High vessel water level resulted causing turbine trip and scram on stop valve closure.	G	3	Reactor Coolant (CH)	Valves	D1.2
27)	9/12	4	50		Reactor scram on low condenser vacuum.	A	3	Steam & Power (HC)	Heat Exchangers	D2.5
28)	9/23	3	70		Reactor shutdown by control rod insertion in order to repair oil leak in main transformer.	B	1	Electric Power (EB)	Transformers	N1.1.4
29)	9/24	80	0		Shutdown by insertion of control rods for main transformer oil leak repair.	B	1	Electric Power (EB)	Transformers	N1.1.4
30)	10/1	28	40	LTR: 10/10/70	Reactor shutdown by control rod insertion following HPCI and Electromatic Relief valve failures.	A	1	Engineered Safety Features (SF-C)	Valves	N1.1.4
31)	10/10	4	90		Turbine trip on moisture separator tank high level caused reactor scram on stop valve closure.	A	3	Reactor Coolant (CC)	Heat Exchangers	D2.3
32)	10/13	244	90		Anticipated scram following manual turbine trip as part of STP #18. Unit remained shutdown to repair condenser tube leak.	A	2	Steam & Power (HC)	Heat Exchangers	N3.1

Table A1.1 (Continued)

No.	Date (1970)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
33)	11/14	110	48		Reactor shutdown by control rod insertion to repair reactor feed pumps and replace 5 control rod drives.	B	1	Reactor Coolant (CH) & Reactor (RB)	Pumps & Control Rods	N1.1.4
34)	11/19	6	NA	LTR: 12/18/70	Reactor shutdown by control rod insertion to allow drywell entry to investigate Electromatic Relief Valve failure and a suspected leak in the drywell.	B	1	Engineered Safety Features (SF-C)	Valves	D6.1
35)	12/4	44	50	LTR: 12/11/70	Reactor shutdown by control rod insertion following MSIV failures (failed to close) during surveillance testing.	A	1	Reactor Coolant (CD)	Valves	N1.1.4
36)	12/7	465	10		Main transformer failure caused a generator and turbine trip followed by a reactor scram on turbine stop valve closure.	A	3	Electric Power (EB)	Transformers	D2.3
37)	12/31	13	NA	LTR: 1/8/71	Repair of air line leak in drywell.	B	1	Auxiliary Process (PA)	Pipes, Fittings	N1.1.4

Table A1.2 Forced Outages and Power Reductions

No.	Date (1971)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1/1	43	0		Continuation of outage 12/31/70.	B	4	Reactor Coolant (CD)	Pipes, Fittings	N1.1.4
2)	1/9	19	60		Repair turbine steam piping leaks and drywell drain sump pump.	B	1	Reactor Coolant (CC)	Pipes, Fittings	N3.1
3)	1/10	14	NA		Reactor scram from IRM Hi-Hi signal due to spike in feedwater flow while placing a feedwater regulating valve in manual.	G	3	Reactor Coolant (CH)	Valves	D1.2
4)	1/22	33	50	LTR: 1/29/71	Spurious high water level signal in moisture separator tank. Following scram upon receipt of a primary coolant isolation signal one MSIV did not close.	A	3	Reactor Coolant (CC)	Instrumentation & Controls	N2.4
5)	2/13	2	90		Turbine balancing.	B	1	Steam & Power (HA)	Turbines	N1.1.4
6)	2/13	27	0		Reactor scram due to incorrect setting of 600 psig interlock setting.	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N6.1
7)	5/28	34	10	LTR: 6/4/71	Drywell entry to locate and eliminate leakage in drywell instrument air system. Also, repair of HPCI return line accomplished.	B	1	Auxiliary Process (PA)	Pipes, Fittings	N1.1.4
8)	6/7	7	70		Turbine trip initiated by spurious turbine moisture separator high water level signal. Closure of turbine stop valves resulted in scram.	A	3	Reactor Coolant (CC)	Heat Exchangers	D2.3
9)	6/7	122	20		Turbine moisture separator modification.	B	1	Reactor Coolant (CC)	Heat Exchangers	N1.2.1

Table A1.2 (Continued)

No.	Date (1971)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
10)	6/14	5	24		Turbine trip due to moisture separator tank "high level" closure of turbine stop valve resulted in a scram.	A	3	Reactor Coolant (CC)	Heat Exchangers	D2.3
11)	6/14	7	0		Reactor scrammed due to low condenser vacuum at reactor pressure 600 psig.	A	3	Steam & Power (HC)	Heat Exchangers	D2.5
12)	6/14	7	0		Shutdown to investigate scram discharge volume high levels.	B	1	Reactor (RB)	Accumulators	N1.1.4
13)	7/2	6	65		During investigation of oscillations in #2 turbine control valve, an oscilloscope used in the checkout caused a ground and the control valves closed resulting in APRM Hi-Hi scram.	G	3	Steam & Power (HA)	Valves	D2.3
14)	7/13	6	64		Drywell pneumatic air supply system was being isolated for repairs. During this operation the backup air supply was inadvertently isolated, MSIV closure occurred and reactor scrammed.	G	3	Auxiliary Process (PA)	Blowers	D2.4
15)	7/14	3	25		Moisture Separator drain tank high level trip occurred while resetting the moisture separator normal drain valves to the high pressure heaters.	G	3	Reactor Coolant (CC)	Valves	N6.1

Table A1.3 Forced Outages and Power Reductions

No.	Date (1972)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	5/9	2	0		Trouble with control rod insert/ withdraw switch.	A	1	Reactor (RB)	Instrumentation & Controls	N2.1
2)	5/10	1	15		IRM Hi-Hi signal - trouble shooting EHC system - opened and closed bypass valves.	G	3	Steam & Power (HA)	Valves	N6.3
3)	5/23	4	30		Group I Isolation resulting from trouble shooting #3 control valve.	G	3	Steam & Power (HA)	Valves	D2.4
4)	6/12	42	95		Condenser tube leak.	A	1	Steam & Power (HC)	Heat Exchangers	N3.1
5)	6/16	10	75		Spurious turbine trip.	A	3	Steam & Power (HA)	Turbines	D2.3
6)	6/22	10	95	LTR: 6/25/72	Low EHC oil pressure sensors at turbine control valves caused "load reject" relays to operate. EHC oil pressure oscillation during turbine surveillance.	A	3	Steam & Power (HA)	Valves	D2.2
7)	6/25	13	90		Steam leak in x-area. Reactor scrams from Group I isolation. Broken fitting on test pilot bled air off valve operator causing MSIV to close.	A	3	Steam & Power (HB)	Valves	D2.4
8)	7/17	98	75		Seal leak on 2B recirculation pump.	B	1	Reactor Coolant (CB)	Pumps	N3.1
9)	8/6	3	90	LTR: 8/23/72	Low EHC oil pressure caused a load reject scram.	A	3	Steam & Power (HA)	Valves	D2.2

Table A1.2 (Continued)

No.	Date (1971)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
16)	7/19	3	50		Turbine stop valve closure caused by turbine trip resulting from spurious reactor high water level signal traced to pressure fluctuation in instrument line header caused when another instrument on same header was placed in service following a surveillance test.	G	3	Auxiliary Process (PA)	Instrumentation & Controls	D2.3
17)	7/20	4	35	LTR: 9/8/71	The reactor scrammed on APRM Hi-Hi signal following a transient initiated by a partial scram. The partial scram was initiated during a surveillance test when a spurious safety system (main steam line radiation monitor) trip occurred at the same time the safety system was being reset, resulting in a partial insertion of control rods. The resulting step decrease in load (708-320 MWe) caused a pressure transient and reactor scrammed on high flux.	B	3	Reactor Coolant (CC)	Instrumentation & Controls	N2.4
18)	7/23	50	50		Repair of leak in drywell pneumatic supply system.	B	1	Auxiliary Process (PA)	Pipes, Fittings	N1.1.4
19)	7/25	4	0		Loss of M/G set when Bus 29 accidentally tripped (loss of power to M/G set). Reactor pressure was below 600 prig, MSIV's closed and condenser vacuum below 23" Hg.	A	3	Reactor Coolant (CB)	Electrical Conductors	N1.1.4
20)	9/30	90	50	LTR: 11/1/71	Investigate effects of a potential water hammer in shutdown cooling piping during surveillance test of system's isolation valves.	B	1	Reactor Coolant (CD)	Pipes, Fittings	N1.1.4

Table A1.3 (Continued)

No.	Date (1972)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
10)	8/17	10	90		APRM high flux sensor trip caused by pressure spike generated when turbine valves closed rapidly.	A	3	Steam & Power (HA)	Valves	D2.3
11)	8/20	7	95		Hi-Hi moisture separator water level trip switch had a bare wire giving a ground and a turbine trip.	A	3	Reactor Coolant (CC)	Instrumentation & Controls	D2.3
12)	9/7	312	75		1B main steam line flow restrictor problems.	A	1	Steam & Power (HB)	Pipes, Fittings	N1.1.3
13)	10/13	7	75		Place weights on turbine for balancing.	B	1	Steam & Power (HA)	Turbines	N1.1.4
14)	10/14	541	50		Main steam line flow restrictor replacement.	B	1	Steam & Power (HB)	Pipes, Fittings	N1.1.3
15)	11/16	1	90		Valve 2301-4 failed to open.	B	1	Engineered Safety Features (SF-C)	Valves	N1.2.4
16)	11/16	10	90		Turbine balancing problems.	A	1	Steam & Power (HA)	Turbines	N1.1.4
17)	11/27	5	90		Manual scram initiated because rods drifted in due to loss of instrument air.	A	3	Auxiliary Process (PA)	Control Rods	N1.1.4
18)	12/2	7	95		Spurious generator trip on loss of field.	A	3	Steam & Power (HA)	Generators	D2.2

Table A1.3 (Continued)

No.	Date (1972)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
19)	12/14	13	90		Oil leak in EHC system.	B	1	Steam & Power (HA)	Valves	N1.1.4
20)	12/22	194	90		Replacing main steam line drain valves and replacing mechanical seals on "A" recirculation pump.	B	1	Reactor Coolant (CC) & (CB)	Valves and Pumps	N3.1
21)	12/31	2	0		High IRM flux.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N2.4

Table A1.4 Forced Outages and Power Reductions

No.	Date (1973)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1/13	20	90		Replace servomotors for intercept and control valves.	B	1	Steam & Power (HA)	Motors	N1.1.4
2)	1/18	20	50		Low reactor water level caused by feedwater regulator valve problem.	A	3	Reactor Coolant (CH)	Valves	D2.7
3)	2/19	5	95		Low reactor water level caused by feedwater regulator valve problems.	A	3	Reactor Coolant (CH)	Valves	D2.7
4)	3/25	211	90	LTR: 4/5/73	Modification to off gas system. During this outage an explosion occurred in the off gas system caused by welder's torch igniting hydrogen in system.	B	1	Radioactive Waste Management (MB)	Pipes, Fittings	N1.1.1
5)	4/28	44	70		Repair steam leaks in x-area and work on maximum recycle system, reboiler main steam and condenser tie in.	B	1	Reactor Coolant (CC)	Pipes, Fittings	N3.1
6)	4/30	6	45		Premature switching from "startup" to "run" mode caused a Group I isolation.	G	3	Reactor (RB)	Control Rods	N6.1
7)	6/3	9	90		Replace PMG on turbine generator.	B	1	Steam & Power (HA)	Generators	N1.1.4
8)	6/5	31	0		PMG problems on turbine generator discovered after initial role.	B	1	Steam & Power (HA)	Generators	N1.1.4
9)	6/16	40	90		Repair leaks on 2B recirculation pump.	B	1	Reactor Coolant (CB)	Pumps	N3.1

Table A1.4 (Continued)

No.	Date (1973)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
10)	6/18	2	50		Rods inserted to bring reactor drain to reactor H_2O temperature ΔT within 140°F.	G	1	Reactor Coolant (CA)	Vessels, Pressure	N6.1
11)	8/2	264	55	LTR: 10/15/73	Inspect snubbers (regulatory requirement).	D	1	Engineered Features (SH)	Shock Suppressors & Supports	N8.0
12)	9/15	10	90		Low oil level alarm on 2A recirculation pump.	A	3	Reactor Coolant (CB)	Pumps	N1.1.4
13)	9/29	4	90		APRM Hi-Hi caused by "2B" recirculation pump flow spike.	A	3	Reactor Coolant (CB)	Pumps	D1.2
14)	10/5	68	80	LTR: 10/15/73	Inspect snubbers (regulatory requirement).	D	1	Engineered Safety Features (SH)	Shock Suppressors & Supports	N8.0
15)	10/19	5	90		MSIV closure due to drywell pneumatic supply valve being closed.	G	3	Auxiliary Process (PA)	Valves	D2.4
16)	11/11	62	90		Repair turbine shaft driven oil pump bearing and permanent magnet generator (PMG).	A	3	Steam & Power (HA)	Pumps	N1.1.4
17)	11/14	2	0		Low reactor water level.	A	3	Reactor Coolant (CH)	Pumps	D2.7
18)	11/27	5	90		Low reactor water level caused by feed pump trip.	A	3	Reactor Coolant (CH)	Pumps	D2.7

Table A1.5 Forced Outages and Power Reductions

No.	Date (1974)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	2/12	144	80	LTR: 2/20/74	Feedwater check valve seal ring and recirculator pump seal leaked excessively.	A	1	Reactor Coolant (CH) & (CB)	Valves and Pumps	N3.1
2)	2/18	329	0		Turbine generator turning gear was damaged (was not engaged properly).	G	1	Steam & Power (HA)	Turbines	N6.1
3)	3/9	57	64		Standby liquid control system valve and pilot valve of an electromatic relief valve leaked excessively.	A	1	Auxiliary Process (PC) & Reactor Coolant (CC)	Valves and Valves	N3.1 and N1.1.4
4)	3/16	12	80		Spurious Hi-Hi moisture separator level signal caused a turbine trip.	A	3	Reactor Coolant (CC)	Instrumentation & Control	D2.3
5)	7/27	116	50	LTR: 8/2/74	Leaky containment isolation valves.	A	1	Reactor Coolant (CD)	Valves	N1.1.4
6)	8/3	1	50		Replaced generator reverse power relay.	A	1	Steam & Power (HA)	Relays	N1.1.4
7)	8/22	112	60		Uncoupling problems with control rod drives.	A	1	Reactor (RB)	Control Rod Drives	N1.0
8)	9/1	35	75	LTR: 9/6/74	Cooling water line to condensate booster pump ruptured.	A	1	Reactor Coolant (CH)	Pipes, Fittings	N3.1
9)	9/3	14	25		Generator/Turbine mismatch signal due to valving error.	G	3	Steam & Power (HA)	Valves	N6.1

Table A1.5 (Continued)

No.	Date (1974)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
10)	9/12	580	85	LTR: 9/19/74	Leaks in recirculation system piping.	A	1	Reactor Coolant (CB)	Pipes, Fittings	N3.1
11)	10/8	9	50		Pressure regulator circuitry.	A	1	Reactor Coolant (CC)	Instrumentation & Controls	N2.0
12)	10/19	99	70	AO:74-52 10/25/74	Bad recirculation pump seal.	A	1	Reactor Coolant (CB)	Pumps	N3.1

Table A1.6 Forced Outages and Power Reductions

No.	Date (1975)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	5/23	66	50	AO:75-30 LTR: 5/30/75	Problems with electromatic relief valve and MSIV. First some of the relief valves required routine maintenance, i.e., cleaned, capped and new gaskets. Second, several were inoperable due to mechanical problems (too much clearance between cap screw and pilot valve stem). Thirdly, one MSIV inadvertantly closed. Fourth, one relief valve failed to open.	B	3	Reactor Coolant (CC)	Valves	D2.4
2)	6/13	85	50		Problem with electromatic relief valves. Cleaned and capped seats of several. One completely failed to operate.	B	1	Reactor Coolant (CC)	Valves	N1.1.4
3)	7/8	15	75		Instrument mechanic scrammed unit while surveillance testing.	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N6.3
4)	8/2	10	97		EHC backup pressure regulator failed.	A	1	Steam & Power (HA)	Instrumentation & Controls	N1.1.4
5)	9/12	0	90-65		Power reduction. Turbine control valve oscillations as load increased.	A	5	Steam & Power (HA)	Valves	N1.1.4
6)	9/20	31	80		Turbine control valve problems.	A	1	Steam & Power (HA)	Valves	N1.1.4
7)	9/24	93	65		Snubber inspection and other maintenance.	B	2	Engineered Safety Features (SH)	Shock Suppressors	N1.1.4
8)	9/29	48	10	AO:75-46	Nitrogen bypass valve left open causing high drywell pressure.	G	3	Engineered Safety (SE)	Valves	N6.1

Table A1.6 (Continued)

No.	Date (1975)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
9)	10/8	126	70	AO:75-48	Nitrogen inerting pipe crack.	A	1	Engineered Safety (SE)	Pipes, Fittings	N1.1.4
10)	10/16	41	65		H.P. turbine inlet steam leak.	A	1	Steam & Power (HA)	Pipes, Fittings	N3.1
11)	10/25	8	90		EHC oil leak.	A	1	Steam & Power (HA)	Pipes, Fittings	N1.1.4
12)	11/15	24	90		Turbine EHC oil leak.	A	1	Steam & Power (HA)	Pipes, Fittings	N1.1.4
13)	11/16	5	25		EHC oil leak.	B	1	Steam & Power (HA)	Pipes, Fittings	N1.1.4
14)	11/22	10	80		Turbine control valve problems Also oil leak in recirculation pump.	A	1	Steam & Power (HA)	Valves	N1.1.4
15)	11/28	26	75		Leakage in drywell pneumatic system.	A	1	Auxiliary Process (PA)	Pipes, Fittings	N1.1.4

Table A1.7 Forced Outages and Power Reductions

No.	Date (1976)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1/2	8	93	RO:76-02	Steam leaks from #3 and #4 turbine control valves. Outage due to condenser low vacuum.	A	3	Steam & Power (HA)	Valves	D2.5
2)	2/13	15	98		Recirculation system valve 202-7A packing leak.	A	2	Reactor Coolant (CB)	Valves	N3.1
3)	2/18	19	95		The 2A feedwater regulating valve stem broke and valve disc drifted shut. Scram from low water level.	A	3	Reactor Coolant (CH)	Valves	D2.7
4)	5/19	121	0	RO:76-31	Startup delayed because of accidental boron injection. Valving procedures were in error.	G	4	Auxiliary Process (PC)	Valves	N7.0
5)	5/25	36	0	RO:76-34	During startup, the "A" Target Rock safety relief valve stuck open.	A	4	Reactor Coolant (CC)	Valves	D6.1
6)	6/7	0	98-25	RO:76-40	Power reduction. Both air ejector rupture diaphrams ruptured.	A	5	Steam & Power (HC)	Valves	N1.1.4
7)	6/25	45	90		Repaired "D" TIP indexes and replaced SJAE rupture discs.	B	1	Instrumentation & Controls (IE) and Steam Power (HC)	Mechanical Function Units & Valves	N1.1.4
8)	6/27	45	10		Moisture separator high level caused a turbine trip.	A	3	Steam & Power (HB)	Heat Exchangers	D2.3
9)	8/21	24	100		EHC oil leak.	B	1	Steam & Power (HA)	Valves	N1.1.4

Table A1.7 (Continued)

No.	Date (1976)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
10)	9/16	0	100-56		Power reduction. Recirculation pumps lower lube oil level was low.	B	5	Reactor Coolant (CB)	Pumps	N1.1.4
11)	9/19	6	50	AO:76-57	Blown fuse in B feedwater regulating valve control circuit caused high water level scram.	A	3	Reactor Coolant (CH)	Circuit closers/ Interrupters	D1.2
12)	10/17	0	100-50		Power reduction. Spurious "A" recirculation pump trip caused power reduction.	A	5	Reactor Coolant (CH)	Pumps	D3.1
13)	11/3	0	100-60		Power reduction. Switching error caused temporary loss of a recirculation pump forcing a power reduction.	G	5	Reactor Coolant (CH)	Pumps	D3.1
14)	11/13	90	100		Repaired "B" TIP indexer, drywell vent valve, and 3/4 inch drain line.	B	2	Instrumentation & Controls (IE)	Mechanical Function Units	N1.1.4
15)	12/18	11	94	RO:76-70	Switchyard voltage transient caused instrument spikes and scram.	A	3	Electric Power (EB)	Electrical Conductors	N1.1.4
16)	12/29	18	100		EHC oil leak.	A	2	Steam & Power (HA)	Valves	N1.1.4

Table A1.8 Forced Outages and Power Reductions

No.	Date (1977)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	2/5	37	98		Low level oil condition on lower level bearing of "A" recirculating water pump.	A	1	Reactor Coolant (CB)	Pumps	N1.1.4
2)	3/16	105	93		Tours inspection, snubber inspection, recirculation water pump seal replacement.	B	1	Reactor Coolant (CB)	Pumps	N3.1
3)	4/1	44	98	RO:77-13	Unidentified leakage in drywell which turned out to be a water leak in a chemical test flange in the bypass loop on "B" recirculation loop.	A	1	Reactor Coolant (CB)	Pipes, Fittings	N3.1
4)	4/5	0	70-50		Power reduction: "A" recirculation pump upper lube oil alarm.	A	5	Reactor Coolant (CB)	Pumps	N1.1.4
5)	4/7	0	70-37		Power reduction. "A" recirculation pump bearing alarm.	A	5	Reactor Coolant (CB)	Pumps	N1.1.4
6)	4/13	0	75-50		Power reduction. Recirculation pump lube oil Hi/Lo alarm. Dry-well entry to remedy problem.	A	5	Reactor Coolant (CB)	Pumps	N1.1.4
7)	4/17	14	70		Packing leak on main steam supply stop valve to "A" SJAE.	A	3	Steam & Power (HB)	Valve	N3.1
8)	4/17	24	10		Electrical feed failure to essential service bus.	A	3	Electric Power (EB)	Electrical Conductors	N1.1.4
9)	5/24	0	80-50		Power reduction. M/G oil pump leak.	A	5	Reactor Coolant (CB)	Generator	N1.1.4

Table A1.8 (Continued)

No.	Date (1977)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
10)	6/4	23	75		The sustained high pressure trip switch was being changed out. When switch was valved back into sensing line, which is common to reactor low level trip switch, unit scrammed on reactor water low level trip.	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls	D2.7
11)	6/14	0	60-25		Power reduction. Rapidly decreasing condenser vacuum.	A	5	Steam & Power (HC)	Heat Exchanger	D2.5
12)	7/9	15	80		A dead weight accidentally dropped against reactor level instrumentation during routine reactor high pressure instrument surveillance.	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N6.3
13)	7/23	0	70-10		Power reduction. Low oil level on recirculation pump. Drywell entry to correct problem.	A	5	Reactor Coolant (CB)	Pumps	N1.1.4
14)	8/11	0	60-25		Power reduction. Excessive steam leak in the x-area.	A	5	Steam & Power (HB)		N3.1
15)	9/7	16	50		Loss of air to MSIV's due to valve diaphram failure. MSIV's failed closed.	A	3	Auxiliary Process (PA)	Valves	D2.4
16)	11/23	428	0		Continuation of refueling outage except now called forced maintenance outage since all maintenance work not completed yet.	B	4	Unknown	Unknown	N1.0

Table A1.9 Forced Outages and Power Reductions

No.	Date (1978)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1/1	19	85		Turbine stop valve - faulty switch in test circuit.	A	3	Steam & Power (HA)	Instrumentation & Controls	D2.3
2)	2/9	13	100		Turbine trip on stop valve closure.	A	3	Steam & Power (HA)	Turbines	D2.3
3)	2/10	29	0		Turbine trip stop valve closure.	A	3	Steam & Power (HA)	Turbines	D2.3
4)	2/25	50	100		Hi MSIV pilot valve temperature. Maintenance error.	G	1	Reactor Coolant (CD)	Valve	N6.3
5)	4/27	27	100		A surveillance procedure did not make reference to a crosstie between breakers in units 2&3. Opening this breaker caused undervoltage on Unit 2 resulting in a scram.	F	3	Electric Power (EB)	Circuit Closers/ Interrupters	N7.0
6)	5/20	76	95		Condenser tube leakage.	A	1	Steam & Power (HC)	Heat Exchangers	N1.1.4
7)	6/25	27	95		Generator load rejection due to a phase mismatch during a severe storm.	H	3	Electric Power (EA)	Electrical Conductors	D2.2
8)	7/28	20	90		Contractor crew dug into an instrument air line to the filter building.	H	3	Auxiliary Process (PA)	Pipes, Fittings	N9.1
9)	7/29	10	10		Operator used a walkie talkie in the auxiliary electric room and tripped the turbine 12% overspeed circuit.	G	3	Instrumentation & Controls (FE)	Instrumentation & Controls	D2.3
10)	8/1	199	60		Bonnet leak on "B" recirculation pump discharge valve.	A	1	Reactor Coolant (CB)	Valves	N3.1

Table A1.9 (Continued)

No.	Date (19)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
11)	9/13	22	90		X-area Hi temperature coolers tripped.	A	3	Other Auxiliary (AA)	Heat Exchangers	N1.1.4
12)	11/6	0	100-25		Power Reduction. Adjust MSIV closure times.	B	5	Reactor Coolant (CD)	Valves	N2.0
13)	12/9	20	90		Recirculation MG set brush replacement.	A	1	Reactor Coolant (CB)	Generators	N1.1.4
14)	12/18	0	90-30		Power Reduction. Lube oil pump trip caused trip of "A" reactor recirculation pump MG set.	A	5	Reactor Coolant (CB)	Pumps	N1.1.4

Table A1.10 Forced Outages and Power Reductions

No.	Date (1979)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1/15	0	90-40		Power reduction. Add oil to "A" recirculator pump.	A	5	Reactor Coolant (CB)	Pumps	N1.1.4
2)	2/3	49	80		Loss of secondary containment due to overpressurization of reactor building as a result of the loss of exhaust fans.	A	1	Other Auxillary (AA)	Blowers	N1.1.4
3)	2/8	16	80		Trip of both scram channels while performing instrumentation surveillance.	G	3	Instrumentation & Controls (IA)	I&C	N6.3
4)	5/5	44	5		"D" TIP machine stuck in index position #2.	A	1	Instrumentation & Controls (IE)	Mechanical Function Units	N1.1.4
5)	5/7	29	5		"D" TIP machine stuck in position #6.	A	1	Instrumentation & Controls (IE)	Mechanical Function Units	N1.1.4
6)	5/9	0	?-10		Steam leak in turbine hood.	A	5	Steam & Power (HA)	Turbines	N3.1
7)	6/6	0	100-50	LER-79-41	Power reduction. "B" reactor recirculation pump MG set tripped when a transformer in the generator field control failed resulting in a load reduction.	A	5	Reactor Coolant (CB)	Transformers	D2.2
8)	6/12	12	100		Inadvertent closure of MSIV.	G	3	Reactor Coolant (CD)	Instrumentation & Controls	D2.4
9)	7/27	0	95-50		Power reduction. High dry well floor drain leakage.	A	5	Reactor Coolant (CB)	Valves	N3.1
10)	7/31	49	60		Repair packing leak on core spray manual stop.	A	1	Engineered Safety (SF-D)	Valves	N3.1

Table A1.10 (Continued)

No.	Date (1979)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/NSIC(N) Event Category
11)	10/13	183	100		Hanger and anchor bolt inspection (regulatory requirement).	D	1	Engineered Safety (SH)	Shock Suppressors	N8.0
12)	10/20	19	10		Moisture separator drain tank high level	A	3	Reactor Coolant (CC)	Instrumentation & Controls	N1.1.4
13)	11/10	15	100		Feedwater pump tripped due to low suction pressure trip introduced.	G	3	Reactor Coolant (CH)	Instrumentation & Controls	D2.7
14)	11/30	54	100		Repair leak on moisture separator line.	B	1	Reactor Coolant (CC)	Pipes, Fittings	N3.1

Table A1.11 Forced Outages and Power Reductions

No.	Date (1980)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	1/30	14	90		Unit inadvertently manually scrammed upon receiving a half-scram signal on RPS channel B.	G	2	Instrumentation & Controls (IA)	Instrumentation & Controls	N6.3
2)	2/3	11	90		Rx scram on low vacuum while transferring max recycle reboiler relief discharge from unit 3 condenser to unit 2 condenser.	A	3	Steam & Power (HC)	Heat Exchangers	D2.5
3)	5/12	10	90	LER:80-17	Rx scram on MSL Hi Rad (spurious signal) resulting in Group I isolation.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N2.4
4)	5/12	145	0		Replaced "B" recirculation pump seals.	A	1	Reactor Coolant (CB)	Pumps	N3.1
5)	5/22	11	80		Rx low level. (Instrument rack jarred inadvertently).	G	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N6.3
6)	7/26	63	75		Manual and automatic scram per IE 80-17 to verify CRD system function properly (Regulatory requirement).	D	2	Reactor (RB)	Control Rod Drives	N8.0
7)	9/23	78	100		Low vacuum. Excessive air inleakage into condenser from "C".	A	3	Steam & Power (HC)	Heat Exchangers	D2.5
8)	10/5	8	50		Place off gas system back in service.	B	2	Gaseous Radioactive Waste Management (MB)	Unknown	N1.0

Table A1.11 (Continued)

No.	Date (19 80)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
9)	10/9	84	50		EHC pump electrical malfunction caused turbine stop valve closure. Unable to bring level up due to inoperability of HPCI system.	A	3	Steam & Power (HA)	Pumps	D2.3
10)	11/20	18	50		2B CRD pump tripped and failed to restart.	A	2	Reactor (RB)	Pumps	N1.1.4
11)	11/24	28	50	LER:80-44	Group I isolation while doing surveillance test-MSL Hi flow.	G	3	Reactor Coolant (CC)	Instrumentation & Controls	D2.4
12)	12/02	73	50		Turbine trip due to moisture in turbine vibration meter.	A	3	Steam & Power (HA)	Instrumentation & Controls	D2.3
13)	12/11	45	50		Scram discharge volume Hi-Hi alarm would not reset.	A	2	Reactor (RB)	Instrumentation & Controls	N1.1.4

Table A1.12 1981 Forced Shutdowns and Power Reductions for Dresden 2

No.	Date (1981)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
1)	5/12	12			Generator ground-manual turbine trip.	A	1	Steam & Power (HA)	Generators	D2.3
2)	6/13	50			Lightning caused in-house electrical problems. Being investigated by SEED.	H	3	Electric Power (ED)	Instrumentation & Controls	N9.2
3)	6/30	37			Recirculation pump tripped while operator adjusting flow. Reactor manually shutdown to repair.	A	1	Reactor Coolant (CB)	Pumps	D3.1
4)	7/15	37			Reactor manually tripped to repair 2B recirculation pump.	A	2	Reactor Coolant (CB)	Pumps	N1.1.4
5)	8/15	32			Maintenance outage to repair 2B recirculation oil pump. (Reactor scram after unit offline because of IRMs Hi-Hi condition.)	A	3	Reactor Coolant (CB)	Pumps	N1.1.4
6)	9/21	39			Mechanical problem in circuitry of turbine stop valves.	A	3	Steam & Power (HA)	Instrumentation & Controls	N2.0
7)	11/3	99	100		Replaced recirculation pump seal.	A	1	Reactor Coolant (CB)	Pumps	N3.1
8)	12/3	17	100		Scrammed on low reactor water level due to instrumentation failure. Calibrate instruments.	A	3	Instrumentation & Controls (IA)	Instrumentation & Controls	N2.0

Table A1.12 (Continued)

No.	Date (1981)	Duration (Hrs)	Power (%)	Reportable Event	Description	Cause	Shutdown Method	System Involved	Component Involved	DBE(D)/ NSIC(N) Event Category
9)	12/12	75	100		Reduced power and removed turbine from grid to repair leak in x-area. Placed generator back in operation. While increasing power scrammed on low reactor water level.	A	9	Reactor Coolant (CH)	Unknown	D2.7
10)	12/24	34		LER: 81-78	While performing safety relief valve operability surveillance "2B" SRV failed to open. Due to HPCI being inoperable, reactor was shut down. The valve actuator was repaired.	A	1	Reactor Coolant (CC)	Valves	N1.1.4

Appendix A: Dresden 2

Part 2. Reportable Event Coding Sheets

Table A2.1 Coding Sheet for Reportable Events for Dresden 2 - 1970

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
70-01	41938	1/26	2/6	D	EG	N,F	-	C	ED	H	N	Reactor was shutdown; an error occurred during testing.
70-02	-	1/26	2/6	C	EG	N	-	C	AQ	G	C1	The 2/3 swing diesel was inoperable.
70-03			4/17	B	SF-C	OO	-	C	AZ	G	C1	HPCI pump supply valve inadvertently open.
70-04	47290	4/23	5/22	C	SF-A	QQ,RR	-	C	BA	D	N	One of five electromatic relief valves failed to open due to misadjusted linkage.
70-05	47289	5/8	5/18	C	CD	QQ,OO	-	C	AG,BB	D	S2	The 1C, 2C, 2A and 2D MSIV's failed to close during testing.
70-06		-	5/14	B	RB	J	-	C	BI	D	C7	One control rod insertion time exceeded limit during startup testing.
70-07	46459	5/28 5/29	6/2	D	SF-C	Z,GG	-	C	HH	A,H	C1	Water hammer due to improper valve arrangement caused HPCI snubber failure.
70-08	46746	5/29	6/8	C	SF-A	QQ,RR	-	C	AZ	D	N	One of five electromatic relief valves failed to open due to misadjusted linkage.
70-09	47822	6/3	7/23	B	CD	OO	-	C	BI	D	N	Two MSIV's in different steam lines closed at times slower than Tech Specs.

Table A2.1 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
70-10	47814	6/5	7/6	B	RA	OO	–	B	EG	D	S5	The Dresden 2 blowdown incident.
70-11	47291	–	6/26	B	RB	J	–	C	BI	D	N	Control rod scram times increasing.
70-12	00077	–	8/25	–	RB	OO	–	C	HL,BB	D	N	Sodium pentiborate crystals prevented the relief valve from closing.
70-13	56054	–	9/10	B	SF-C	Y	–	–	BI	B	C1	Wrong size criteria in lubrication line.
70-14	57236	9/15	10/16	B	EC	C	T	B	ED	C	C1	A 125 V DC battery discharged and the inverter tripped.
70-15	60226	9/27	12/21	–	MA	JJ	–	B	OD	A	N	Radioactive limit exceeded in above ground storage tank.
70-16	57235	10/1	10/9	D	CC,CA	OO	–	C	BA,AB	D,G	C1	Tight packing caused steam supply valve to fail. Worn solenoid caused electromatic valve to fail.
70-17	57053	10/15	10/15	D	PC,MA	V	–	B	OD,OK	A,H	C3	Small release of radioactive liquid. The radwaste concentrator leaked.

Table A2.1 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
70-18	56981	10/15	10/22	D	PC,MA	V	-	B	OD,OK	A,H	C3	Small release of radioactive liquid to the environment.
70-19	-	10/20	11/30	D	PA	-	-	C	BU	A	N	NaB_5 11.2% instead of 12% by weight.
70-20	-	10/27	11/27	B	CD	OO	-	C	AW	D	N	MSIV 1C closed too quickly.
70-21	58012	10/27	11/27	B	RB	J	-	C	BI	D	N	Nine control rods exceed insertion time.
70-22	58012	10/28	11/27	B	RB	J	-	C	BI	I	N	Two control rods exceeded scram insertion times by 7 sec.
70-23	59931	11/2	12/2	B	EE	N	-	C	AQ	D	N	Swing diesel generator failed to start.
70-24	60228	11/10	12/10	D	CD	OO	-	B	BL	D	N	One MSIV pilot valve temperature too high.
70-25	59928	11/17	12/18	B	RB	J	-	C	OA	C	N	Control rod failed to insert properly.
70-26	59929	11/19	12/18	B	RB	J	-	C	AG	B	N	Control rod failed to withdraw.
70-27	59930	11/19	12/18	D	SF-A	QQ,RR	-	C	BA	D	N	One electromatic relief valve had failed.

Table A2.1 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
70-28	60227	12/4	12/11	B	CD	OO	-	C	AG,AQ	D	S2	Four MSIV's failed to close.
70-29	60603	12/9 12/21	1/15/71	B	CE	OO,F	T	C	AK,BC	G	N	Three valves failed to operate in 3 systems.
70-30	59599	12/30	1/8/71	B	PB	Z	-	B	AW	D	N	Leaking pipe nipple on MSIV pilot valve.

Table A2.2 Coding Sheet for Reportable Events for Dresden 2 — 1971

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
71-01	59597	1/5	1/12	B	MA	X	–	C	BB,BD	D	N	Damper motor failed.
71-02	59490	1/19	1/29	B	SF-C	DD	E	C	HB	D	N	Square root extractor malfunctioned.
71-03	60602	1/22	1/29	B	CD	OO	–	B	AQ	G	N	One MSIV pilot valve was fouled by oil.
71-04	62111	2/3	3/2	B	MB	E	–	B	CA	D	N	Damper in redundant SBGT system failed.
71-05	–	2/26	3/8	B	SF-B,CF	OO,F	–	C	BA	C	C4	HPCI valve failed to automatically open given LOOP.
71-06	65542	3/2-3	8/9 8/13	C	MB	BB	–	A	OK	A	C3	Release rates out the stacks exceeded limits.
71-07	63148	3/7	4/5	D	EG	N	–	C	AQ	G	C7	Swing DG governor speed controller set incorrectly.
71-08	39240	3/17	2/17/72	B	RC,RA	R,KK	F,L	B	OB	B	N	Asymmetrical flux distribution due to non-uniform core inlet temperature.
71-09	63113	–	3/17	C	SC	Z,OO	G	B	AD,AW	B	N	A broken tube fitting, a leaking diaphragm, and trip instruments were repaired, replaced and adjusted to decrease the number of drywell ventings.

Table A2.2 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
71-10	63114	-	3/17	C	MA	P,H,M	-	-	OA	B,A	N	An investigation of liquid radioactive waste system failures.
71-11	63115	-	3/17	B	IE,IF	-	F,L	B	OB	D	N	Asymmetrical flux distribution due to variations in inlet enthalpy.
71-12	63147	3/27	4/3	D	SF-D	-	T	C	AD	D	N	One core spray valve failed to open automatically.
71-13	63141	3/29	4/29	D	SF-D	GG	-	C	HH	D	C8	Six seismic snubbers broke due to a water hammer.
71-14	63142	4/19	4/29	C	EG	N	-	C	AQ	C	C7	Air motor on Unit 2 DG failed to engage the diesel.
71-15	63266	-	5/7	B	RC	R	-	B	BX	D	C3	Potential fuel pin leakers. Off-gas still below Tech Specs limit.
71-16	63200	-	5/21	D	SA	SS	-	D,I	AC	B	C8	Torus paint scaling, discusses this problem.
71-17	63267	5/27	5/27 6/4	B	SF-C	Z	-	B	AD	H	C8,C3	HPCI test return line ruptured.
71-18	64302	-	6/21	B	SF	F	-	C	BF	D	N	Breaker tripped.
71-19	64825	7/6	7/14	B	EG	N	-	C	BD	D	N	The 2/3 swing diesel tripped due to high crankcase pressure.

Table A2.2 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
71-20	65972	7/20	9/8	B	RB	-	T	C	BI	D	C8	Partial control rod insertion followed a spurious scram signal.
71-21	64647	8/5	8/6	B	SF-C	OO	T	C	BC	D	C1	HPCI turbine failed to reset.
71-22	65545	8/5	8/14	B	SF-C	OO	T	C	BC	D	C1	HPCI turbine stop valve failed to open due to limit switch miscalibration.
71-23	66505	8/7	8/17	D	MA	OO	-	B	OK	H	C3	Improper valving arrangement resulted in an unplanned release.
71-24	64655	8/8	8/8	B	MA	OO	-	B	OK	H	C3	Incorrect valve lineup resulted in an unplanned liquid release.
71-25	65543	8/10	8/10 8/20	B	EE	N	-	C	BD	D	N	Swing diesel failed to start.
71-26	68334	9/28	11/1	B	CB,CF SF	OO,Y,Z	-	C	HH	A	C8	Water hammer damaged a shutdown cooling system pipe.
71-27	67334	10/20	10/28	B	SD	FF,E	-	A	OK	A	N	A Unit 2/3 secondary containment door was opened during maintenance.

Table A2.2 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
71-28	67849	11/16	11/23	B	SF-C	OO	-	C	AL	D	C1	Loose adjusting screw on HPCI control valve.
71-29	69201	11/25	12/23	B	SA	E,DD	T	B	BA	D	N	Vacuum breaker between torus and reactor building failed to remain open.

Table A2.3 Coding Sheet for Reportable Events for Dresden 2 — 1972

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
72-01	55898	2/29	3/6	C	RB	J	T	B	EG	D	N	Refueling interlock failed.
72-02	69702	3/2	3/30	C	EE	N	-	C	BD	D	C7	Air starter motor failed on unit 2 DG.
72-03	70015	3/5	4/3	C	EG	DD,F	-	C	BD	D	N	A LPCI pump failed to start due to breaker problems.
72-04	71708	4/6	5/31	C	CI	OO	-	C	AD	D	N	A broken bolt on a relief valve was found.
72-05	69791	-	4/14	C	SF-C	-	E	C	AD	B	C1	HPCI flow switch found missing paddle & two mounting switches.
72-06	71408	-	5/9	C	SA	JJ	-	C	AC	B	C7	Torus paint problems continue due to paint cracking.
72-07	71710	5/9	5/18	C	SD	OO,FF	-	C	AT,AW	D	N	Containment leak rate exceeds limits at three locations.
72-08	71374	-	5/12	C	CF	-	E	C	AD	B	C1	Shutdown cooling system flow switch's failed.
72-09	71709	5/17	5/25	B	MA	JJ	-	C	BU	H	N	High level of radio-activity in waste tank.
72-10	71714	5/27	6/6	B	IA	OO	T	C	EG	C	N	Turbine stop valve closure failed to initiate reactor trip.

Table A2.3 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
72-11	72420	6/17	6/23	B	SD	OO	-	C	AG	G	N	Containment sample-return isolation valve failed closed.
72-12	72771	7/15	7/27	B	IA	-	T	C	EI	D	N	Set point drift of 2 low pressure scram switches.
72-13	75042	7/15	7/15	B	IA	-	M,T	C	EI	D	N	Two of four reactor vessel high pressure scram switches drifted.
72-14	72770	7/24	7/29	B	EE	N,DD	-	B	AL	G	C7	D.G. failed to run due to loose wire on D.G. water pump.
72-15	72822	7/26	7/26	B	MB	DD	-	B	BF	D	C3	Stack gas sample pump tripped.
72-16	73480	7/26	8/1	B	MB	DD	-	B	BF	H	C8	Operator failed to respond to an alarm when sample pump tripped.
72-17	73342	-	8/3	-	RB	J	-	B	EI	D	C7	Operating history of rod-worth minimizers examined.
72-18	77450	8/14	12/14	B	RB	I	-	B	OC	A	N	Reactor scrammed due to an IRM Hi Hi Flux signal.
72-19	75075 75895	8/29	9/7 10/2	B	CD	OO	-	B	AD	E	C8	Steam flow was restricted when a flow restrictor broke loose.
72-20	75757	9/3	10/3	B	EE	N	P	B	BI	D	C7	The 2/3 swing diesel failed to auto restart.

Table A2.3 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
72-21	75068	9/8	9/18	–	IA	–	T	C	EI	D	N	Set point drift of two low pressure scram switches.
72-22	75603	9/21	9/29	B	IB	–	M	A	EI	D	C7	Two low pressure permissive switches drifted.
72-23	75950	9/29	10/27	B	HA	HH,OO	–	C	AQ	D	N	#1 Turbine control valve malfunctioned.
72-24	76060	10/8	11/6	B	HA	HH,OO	T	C	AC	C	N	#3 Turbine control valve malfunctioned.
72-25	76082	–	10/10	B	SA,SD	FF	–	C	AW	D	N	Four containment penetrations leaked.
72-26	77957	10/13	11/24	B	HJ	JJ	–	B	BE	D	N	Cooling pond failed when a dyke failed and flooded the area.
72-27	75903	10/25	11/3	B	MB	DD	–	B	ED	B	C7	Failure of both stack-monitoring sample pumps.
72-29	76457	10/3	11/27	B	SA,SD	OO	–	C	AW	D	N	Primary containment leaks.
72-30	76405	11/6	11/15	B	SF-C	OO,F	–	C	BA	A	C7	HPCI MOV fails to open.
72-31	76393	11/21	11/21	B	IB	–	M	A	EI	D	C7	Two low pressure permissive switches drifted.
72-28	75909	10/12	11/3	B	CD	–	T	A	EI	D	C7	Set point drift in steam line low pressure switches.

Table A2.4 Coding Sheet for Reportable Events for Dresden 2 — 1973

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
73-01	88082	–	1/17	B	SB	–	I	C	EH	D	N	Set point drift in reactor level switches.
73-02	–	1/23	1/23	B	IA	–	M	C	EI	D	N	Low condenser vacuum setting drifted.
73-03	–	1/23	1/23	B	IB	–	M,T	C	EI	D	N	HPCI steam line detection sensor drifted.
73-04	79413 82275	2/19	3/13	B	RB	J	–	C	CA	D	N	Three control rods uncoupled.
73-05	–	2/20	3/19	B	SF-B	–	M,T	C	EI	D	N	One LPCI pressure switch setscrew missing.
73-06	79604	2/23	3/23	D	WA	OO	–	C	AL	D	N	Failure of RHR heat exchanger outlet valve.
73-07	79602	–	3/7	–	RB,SH-D EE	F	–	–	BG	B	C4	Common cause problems in ESF. Racking out a component prevented the operation of another component.
73-08	80495	3/15	4/12	B	SF-C	OO,X	–	C	BC	G	C1	HPCI isolation valve failed to open.
73-09	80116	3/20	3/29	B	MA	P	–	B	OD	A	N	Activity in radwaste tank exceeds limit.
73-10	80117	3/21 3/23	3/30	B	MA	P	–	B	OD	A	N	Activity in radwaste tank exceeds limit.

Table A2.4 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
73-11	80133	3/27	4/5	C	MB	P	-	B	BY	A	C3	Hydrogen explosion in off-gas system.
73-12	80284	3/30	4/14	B	WA	OO	-	C	BA	D	C7	Failure of service water valve to open automatically.
73-13	80496	4/2	4/13	B	MA	JJ	-	B	OD	A	C7	Activity exceeded in radwaste tanks.
73-14	80133	4/5	4/5	B	MA	JJ	-	A	OD	A	N	Activity in radwaste tank exceeds limit.
73-15	-	4/9	4/9	B	MA	JJ	-	B	OD	A	N	Activity in radwaste tank exceeds limit.
73-16	80494	4/5	4/13	B	CJ	F	-	C	BF	B	C7	An isolation condenser valve breaker was found tripped.
73-17	-	4/19	4/27	C	CD	OO	-	A	BC	A	N	Secondary containment not in effect.
73-18	74828	-	4/27	B	SA	FF	-	B	OK	A	N	Partial loss of secondary containment.
73-19	80729	5/23	6/1 6/17	B	EE	N	-	C	BC	H	N	Incorrect cooling water flow valve lineup to diesel generator during testing.
73-20	81478	6/6	6/15	D	CH	F	P	C	BF	G	C4	Reset spring missing in breaker.
73-21	[illegible]	6/7	6/7	B	CJ	F	-	C	BF	B	N	Isolation cnodenser valve breaker tripped on thermal overload.

Table A2.4 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
73-22	82191	6/18	7/16	B	IB	-	P	A	AL	E	N	A logic switch failed due to installation errors.
73-23	82276	7/9	7/17	B	MA	JJ,P	-	B	OD	A	C7	Activity limit exceeded in floor drain tank.
73-24	82277	-	7/20	B	MA	JJ,P	-	B	OD	A	C7	Activity limit exceeded in floor drain tank.
73-25	83227	7/24	8/22	B	CF	OO	T	C	AP,EE	D	C7	Service water valve failed to open.
73-26	83163	8/2	8/10	B	MA	JJ,P	-	B	OD	A	C7	Activity limit exceeded in floor drain tank.
73-27	83226	8/15	8/24	B	SF-C	OO,F	-	C	AE	D	C1	Two HPCI MOV's fail due to bent relay interlock bars.
73-28	83229	8/15	8/24	B	SF-C	OO	-	C	AE	D	C1	HPCI steam supply valve failed to open due to bent valve stem.
73-29	83843	8/17	9/14	B	SF-B	OO,F	-	B	BC	G	C7	LPCI valve trips during operation.
73-30	83836	8/20	9/18	B	SF-B	OO,F	-	C	AG	D	C7	LPCI valve breaker trips during auto initiation.
73-31	84035	8/21	9/13	B	SF-B	OO	T	C	AD	B	C7	Two LPCI suction valves failed to open.
73-32	84198 84199	9/4	9/21	B	CF	OO	-	B	BC	B	C7	RHR heat exchanger valve failed to open.

Table A2.4 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
73-33	84200	9/15	9/21	D	SD	OO	-	C	AL	G	N	Main steam line drain valve failed to operate due to missing parts.
73-34	85377	10/8	11/5	B	CF	OO,F	-	B	BC	G,B	N	Shutdown cooling valve failed to operate.
73-35	86170	-	10/15	B	SH	GG	-	C	BT,AW	E	N	Snubber oil level was low.
73-36	85319	10/17	10/26	B	SF-C	OO	-	C	EE	G	C1	HPCI valves loose seal-in capability.
73-37	85310 87068	10/19	10/26	B	CJ	OO,F	-	B	EH	A	C7,C1	Isolation condenser valve breaker tripped while valve was opening after scram.
73-38	85569	10/25	11/21	B	SF-B	OO	-	C	AI,AL	D,B	C7	LPCI valve failed to open.
73-39	87047	11/23	11/30	B	SF-C	AA	-	B	ED	B	C1,C7	HPCI DG power supply grounded.
73-40	86995	11/27	11/30	B	SF-C	JJ	-	C	BS	A	N	Torus water level exceeds limit due to test of HPCI system.
73-41	87035	12/3	12/12	B	SF-C	Z	-	-	AQ	G	C1	HPCI steam trap leaks.
73-42	-	12/6	11/27	B	CC	OO	-	B	BC	D	N	Main steam line drain valve failed to operate.
73-43	87034	12/18	12/24	B	CD	-	E	B	EE	D	N	Main steam line hi-flow sensor failed.

Table A2.5 Coding Sheet for Reportable Events for Dresden 2 — 1974

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
	88091	1/1	1/11	B	SA	FF	–	B	AL	B	N	Temperature seal material failed in the containment.
	–	1/8	1/17	–	SB	–	I,T	C	EI	D	N	Core water level sensor drifts.
	88460	1/17	2/6	–	SD	OO	–	C	AW	D	N	Containment leak rate exceeds Tech Specs limit.
	88331	–	2/8	–	SF-D	Z	–	C	AR	D	N	Pin hole leak in core spray low flow line.
	89160	2/12	2/20	–	CA,CH	DD,OO	–	B	AU	D	C8	Recirculation pump seal leaked.
	89161	2/13	2/21	D	SH	GG	–	C	AC	B	N	Twelve shock suppressors inoperable.
	89382	2/23	3/22	–	SF-C	OO	–	C	AN	E	C7	HPCI drain line repaired and a bypass valve was installed backwards.
	89259 92492	3/1	3/8	B	EE	N	–	C	EE	B	C7	Diesel failed when a bus fuse clip failed.
	–	3/9	3/18	D	SA	OO,UU	–	C	AQ	D	N	Drywell isolation valve failed.
	89393	3/11	3/11	D	SD	HH	–	C	AQ	D	N	Crud on solenoid in sump discharge valve.
	89394	3/14	3/21	D	RB	J	–	C	CA	D	N	One C.R. overtraveled when it uncoupled.

Table A2.5 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
	89666	3/24	4/2	D	RB	HH	–	C	BB	D	C7	Turbine stop valve fast closure fails.
	–	3/29	4/25	C	EA	F	–	A	OK	G	N	Maintenance error rendered several systems inoperable.
	90582	4/11	4/18	–	SF-A	OO	–	C	OK	G	C7	Too many vacuum breakers inoperable simultaneously.
	91001	4/25	5/3	D	MB	DD	–	B	AA	D	N	Two stack gas sampling pumps failed.
	91118	4/27	5/6	–	MA	Z	–	B	AR	D	C7	Acid induced leak in radwaste piping.
	91671	5/24,25 26,27	5/31	–	MA	P	–	B	AQ	A	C7	Activity in radwaste surge tank exceeds limit.
	92619	5/27	6/3	–	EE	N	–	C	BI	G	N	Swing DG failed to acquire full load.
	92441	5/28	6/6	–	MA	D	–	B	AQ	A	C7	Activity in above ground tank exceeds limit.
	93686	6/11	6/2	–	SH	GG	–	E	AT	D	C7	8 of 31 snubbers failed.
	93687	6/12	6/19	B	IB	–	T	C	EH	D	N	A high reactor pressure switch's set point drifted.
	93791	6/19	6/24	B	RB	J	–	C	CA	D	N	A control rod became uncoupled.
	94181	6/20,21	6/27	B	MA	JJ,Z	–	B	BU	H	N	Calculations suggest high radwaste concentration.

Table A2.5 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
AO-74-24, 25,26	94730	7/22	7/29	B	SF-D	00	-	C	CA	D	C7	Core spray suction valve failed.
	94799	7/22	7/31	-	SF-B	00	-	C	AL,AM	B,A	C7,C3	LPCI valve failed during test.
AO-74-27	94755	7/20	7/29	B	CE,SH	00	-	C	AE	D	C7	A vacuum breaker failed during test.
AO-74-28	94763	7/20	7/26	B	CC	-	T	C	EH	C	C7	Instrument drift, pressure switch.
AO-74-29	94764	7/20	7/25	B	CC	-	T	C	EH	C	C7	Instrument drift, pressure switch.
AO-74-30	94765	7/24	8/2	B	CC	-	T	C	EH	C	C7	Instrument drift, pressure switch.
AO-74-31	94727	7/29	8/2	-	CC,SH	00	-	C	AR	D	C7	A vacuum breaker leaked.
AO-74-32, 33,34	94728	7/24, 29	8/2	B	SD	Z,00	-	C	AV,AD	A	N	Cracks in isolation valves.
AO-74-35	94914	7/30	8/6	D	CH	Z	-	C	AP	B,E	S9	A four inch crack developed in a ten inch recirculation pipe.
AO-75-36	95098	8/2	8/9	-	RB	J	-	B	CA	D	N	A control rod uncoupled.
AO-74-37	95027	8/1	8/9	-	SF-C	F	-	C	BC	G	N	HPCI tested after maintenance and a breaker valve failed to open.

Table A2.5 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
AO-74-38	95025	8/5	8/14	B	SF,CF	-	I	B	BC	E,G	N	Improper installation of torus level sensor.
AO-74-39	95034	8/10	8/16	-	RB	OO	C	C	EF	D	C8	Failure of RPS signal.
AO-74-40	95380	8/15	8/15	B	SF-C	-	T	C	EI	C	C7	HPCI pressure switch's set point drifted.
AO-74-41	95142	8/23	8/30	C	CH	Z	-	C	AR	B	N	Pin hole leak in feed pump minimum flow line.
AO-74-42	95141	8/24	8/30	C	SH	GG	-	C	AT,BT	D	C8	Eight snubbers were low on oil.
AO-74-43	-	9/1	9/6	B	CH	DD	-	B	AW,AP	B	N	Pipe nipple fails on condensate pipe.
AO-74-46	95593 95880	9/12	9/20	D	CB	Z	-	B	AW	D	S9	75% radial crack in 4" recirc. line.
AO-74-48	96269	9/27	10/7	D	MB	-	-	B	BD	D	N	Train A of SBGT system failed to start.
AO-74-49	96270	9/24	10/8	D	MA	JJ	-	B	BG	A,B	C7, C3	Activity level in waste tank exceeds Tech. Spec. limit.
AO-74-51	96467	10/6	10/15	-	MB	DD,FF	-	B	AC	D	N	Gasket failed in off-gas flow meter pump.
AO-74-52	96535	10/18	10/25	B	CB	OO	-	B	BB	A	N	Recirc. suction valve failed to close.

Table A2.5 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
AO-74-53	96533	10/21	10/31	-	SH	GG	-	B	AC,BC, AW	D	C7	Four of 31 drywell snubbers had low oil levels.
AO-74-54	96534	10/12	10/31	B	CH	Z	-	B	AD	E	C8	Leak from improper weld.
AO-74-56	97140	11/2	11/12	D	RB	J	-	B	CA	D	N	A control rod uncoupled from its drive.
AO-74-57	97498	11/2	11/12	C	IA	E	-	C	GH	D,H	N	An average power range monitor drifted.
AO-74-58	97141	11/2	11/12	D	RB	J	-	B	BE	D	C8	Three control rods insert to position 02.
AO-74-59	97078	11/2	11/12	B	RB	OO	C	C	EE	C	S9	The reactor protection system failed to give an alarm or a scram signal.
AO-74-63	97498	11/11	11/18	C	SF-C	-	E	C	EH	B	C1	Temp probes failed in HPCI.
AO-74-66	97718	11/14	11/22	C	SF-C	OO,S	-	C	AC	D	C1	HPCI CST suction valves failed to open.
AO-74-69	97740	11/20	11/27	C	SF-C	OO,X	-	C	AC	D	C1	HPCI MOV failed due to electrical arcing.
AO-74-70	97739	11/9	11/29	-	MB	UU	-	B	AA	D	C3	Air ejector ruptured in rad gas system. Tech. Spec. limits exceeded.
AO-74-73	98579	12/20	12/20	C	SF-D	Z	-	C	AO,AV	E	S9	Pipe cracks in core spray system.

Table A2.5 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
AO-74-75	98577	12/12	12/20	C	SD	FF	-	C	AV,AU	D	S4	Fourteen bellows leaked, 1 excessively.
AO-74-76	98578	-	12/20	C	SD	Z	-	C	AW	D	C7	Fourteen bellows in the primary containment leaked at the seals.
AO-74-77	98517	12/20	12/20	C	CB	Z	-	C	AV,AO	A,B,D	S9	Four inch recirculation bypass line cracked.
AO-74-78	98584 98715	12/16	12/16	C	CF,SF-B	-	M	C	BZ	B	C4	Pressure switches too sensitive to turbulence.
	98585	12/11	12/19	C	CD	-	U	C	AM	B	C4	Temperature sensors not suited for the purpose required.
AO-74-79	98718	-	12/30	C	CC	OO	-	B	AO	C	N	Non-safety main steam line drain valve failed.
	91658	5/22	5/24	B	IB	-	M	C	EI	C	N	Set point drift in reactor protection systems pressure switch.
	91663	5/22	5/28	B	CC	OO	-	C	EE	E	N	A motor operated main steam line drain valve failed to open.

Table A2.6 Coding Sheet for Reportable Events for Dresden 2 — 1975

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
	93235	-	1/21	B	SF-D	OO,X	-	C	AH	B	N	A core spray motor operated valve failed to open.
AO-75-01	93236	1/10	1/20	C	SF	-	I,T	C	EI	G	N	An ECCS water level switch's set point drifted.
AO-75-01	93278 99307	1/12	1/21	C	SF-D	PP	-	C	BB	D	N	A core spray check valve failed to reset.
AO-75-04	93602	1/20	1/30	C	EE	N	T	C	OK	A	C7, C5	Swing DG failed to start after six hour run due to improper setting of the drop switch.
AO-75-05	93509	1/20	1/30	C	EB	F	P	C	BA	G	N	Load not connected to overcurrent relay after maintenance.
AO-75-07	9350[illegible]	1/22	1/31	C	IA	-	I,T	C	BS,EH	D	N	Loose mounting screws result in core height permissive switch drift.
AO-75-08	93511	1/23	1/31	C	SF-D	PP	-	C	BB	D	N	A core spray check valve failed to reset.
AO-75-09	93514	1/23	1/31	C	SF-C	-	T	C	EE,AQ	E	C1	HPCI valve failed to open completely due to torque switch failure.
AO-75-10	93603	1/25	2/3	C	RB	J	-	A	OK	H	C8	Two adjacent control rods withdrawn.

Table A2.6 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
AO-75-11	93663	1/27	2/6	C	SF-D	Z	-	C	AU	D	C7, C5	Through-wall crack in core spray injection line.
AO-75-12	100046	2/9	2/21	C	SF-D	Z	-	C	AU	D	C7, C5	Through-wall crack in core spray injection line.
AO-75-13	100376	2/28	3/10	C	SF-D	OO	C	A	OK	B	N	A core spray valve failed to open for maintenance due to a circuit lockout arrangement.
AO-75-14	100584	3/3	3/12	C	SE	Z	-	C	AO,AV	E	N	Crack in N_2 purge line weld.
AO-75-15	100583	3/7	3/17	C	SF-C	-	E,T	C	EH	D	C1	Set point drift in HPCI high flow switch.
AO-75-16	101077	3/17	3/27	C	EE	N,X	-	C	BD	D	C7	DG failed to start.
AO-75-17	101076	3/17	3/27	C	CD	-	E,T	C	EH	D	N	Set point drift in main steam line low flow switches.
AO-75-18	101075	3/19	3/27	C	EE	N	P	C	AQ	D	C7	Dirty contacts prevented DG from coming up to voltage.
AO-75-19	101409	3/21	3/31	C	SA	FF	-	B	OI	A	N	Secondary containment air lock doors opened violating Tech Specs.
AO-75-20	101741	4/3	4/15	C	EE	N	C	B	EF	D	C7	DG failed to achieve voltage.

Table A2.6 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
AO-75-21	102292	4/16	4/23	C	MB	FF	-	C	AC	E	N	Gasket leaked in a SBGT charcoal cell.
	101153	-	3/28	C	SF	OO,S	-	B	BA	D	N	One of four high head safety injection valves failed to open.
	101443	-	3/27	C	CD	-	M,T	B	EH	D	N	All 4 main steam line low pressure switches drifted.
AO-75-23	103081	5/11	5/21	C	EE	P,N	-	B	BL	H	C7	DG inadvertently started and overheated during plant maintenance.
AO-75-24	103080	5/13	5/21	C	SA	OO,HH	-	C	BB,AQ	D	N	Drywell equipment drain sump valve failed to close.
AO-75-26	103082	4/15	5/23	C	EE	N,P	-	C	AG	B	C7	DG starting air motor failed.
AO-75-27	103078	4/21	5/27	C	RB	I,J	-	C	OK	H	C3	Control rod withdrawn with personnel within core line of site.
AO-75-28	103079	5/6	5/23	C	SB	OO	-	C	AE,BA	D	N	Two containment cooling water valves failed to open.
AO-75-29	103087	5/19	5/29	B	IB	OO	T	C	BI	G	N	A MSIV 10% closure switch failed.
AO-75-30	103084	5/23	5/30	B	SF	OO	-	B	BA,EI	G	N	Electromatic relief valve failed to open.

Table A2.6 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
AO-75-31	103083	5/22	5/30	B	WB	OO	-	C	AD	D	N	Containment cooling water valve stem broke when a motor operator failed to shut it off.
AO-75-32	103481	5/26	6/3	B	SF	OO	-	C	BA	D	C8	ADS valve failed to open.
AO-75-33	103203	5/26	6/5	-	SA	JJ	-	B	OJ,BT	H	N	Torus water level dropped below limit.
AO-75-34	103202	5/29	6/6	B	SB	OO	-	C	AE	B	N	Containment cooling water valve failed to operate.
AO-75-35	103201	5/26	6/5	B	MB	-	-	B	AZ	G	C8	Explosion in the off-gas system at Dresden 2.
AO-75-36	104046	6/3	6/6	B	IB	OO	T	C	BI	D	N	MSIV 10% closure time excessive.
AO-75-37	103455	6/4	6/13	B	EE	N	-	C	BD	D	C7	DG failed to start.
AO-75-38	103707	6/15	6/25	D	CJ	OO	-	C	AE,BA	H	C1	An isolation condenser valve failed.
AO-75-39	103669	6/12	6/18	B	EE	N	-	C	BD	D	C7	DG failed to start.
AO-75-40	103668	6/11	6/20	B	PA,RB	OO	-	C	EI	D	N	CR 5% scram times exceed limit.
AO-75-41	103667	6/13	6/23	B	SF	OO,FF	-	C	BA	B,D	C1	Two ADS valves failed.

Table A2.6 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
AO-75-42	106345	8/29	9/8	B	EE	N,DD,F	–	C	BF	B	C7	Swing DG failed to run.
AO-75-43	106456	9/11	9/19	B	EE	N,DD,F	–	C	BF	B	C7	Swing DG cooling water pump breaker tripped.
AO-75-44	103085	1/16	5/23	C	RX	O	–	A	AS	A	C8	Roll of tape dropped in the annulus.
AO-75-44	106970	9/23	10/2	B	EE	N,X	–	C	BD	G	C7	DG failed to start due to improper air starter motor maintenance.
AO-75-44	109192	9/23	12/17	B	EE	FF	–	C	BD	A	C7	Wrong O-rings used in diesel generator valve piston.
AO-75-45	–	9/29	10/9	D	SF-C	HH	–	B	EE	D	C1	A cold solder joint caused HPCI's failure to trip.
AO-75-46	106983	9/29	10/9	B	SE	OO	–	B	AY	A,H	C8	Reactor scram due to high drywell pressure.
	104053	–	7/3	C	RB	PP	–	C	AQ	D	N	A CRD check valve leaked.
AO-75-48	106984	10/7	10/9	D	SF-C	OO,HH	–	B	BB	E	C1	HPCI failed to trip during transient.
AO-75-49	107499	10/9	10/17	D	SF-B	–	T	B	EE,EH	B	N	Two switches caused wrong LPCI loop to be selected.
AO-75-50	107500	10/8	10/17	D	CD	–	M,T	C	EH	D	N	Set point drift in main steam line low pressure switches.

Table A2.6 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
	107766	-	10/28	-	SH	GG	-	B	AW	E,G	N	Three of the new snubbers leaked.
AO-75-51	107982	10/31	11/6	B	SF-C	OO	-	C	BS,OJ	H	N	Operator error results in exceeding torus water level limit.
AO-75-52	108557	-	11/20	B	CD	OO	-	C	BC,BI	D	N	A MSIV closed too rapidly.
AO-75-60	103086	-	5/9	C	SA	FF	-	C	AW	D	N	Local leak rates exceeded limits.

Table A2.7 Coding Sheet for Reportable Events for Dresden 2 — 1976

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
RO-76-01	110326	1/1	1/29	D	EA	F	–	B	AL	D	N	A loose cable caused an offsite disconnect bus to fail.
RO-76-02	110325	1/2	1/21	D	HG,HH	H	–	B	AZ	G	N	Loss of condenser vacuum resulted in reactor trip signal.
RO-76-03	110943	1/9	1/26	B	IF	–	F	B	BC	H	N	An in-core tip probe was improperly positioned.
RO-76-04	111604	2/7	2/19	B	IA	–	M	C	BC	D	N	A scram pressure sensor's set point drifted.
RO-76-05	111657	2/17	3/2	B	IE	–	E	C	BC	D	N	Isolation condenser flow sensors drifted.
RO-76-07	112167	3/12	3/29	B	SF	H	–	B	BN	D	N	Drywell to torus ΔP exceeds limit by 1%.
RO-76-09	112166	3/13	3/24	B	SF-C	00,F	–	C	BG	D	C1	HPCI supply valve failed to open.
RO-76-10	112701 112700	3/15	4/12	C	SD	00	–	C	AA,AW	D	N	Drywell isolation valve leaked excessively.
RO-76-11	113279	3/15	4/12	C	SD	00	–	C	AC,AW	D	C7	Leak rate of containment isolation valves exceeded limits.
RO-76-12	112727	3/22	4/5	C	SF-B	F	–	C	BC	G	N	LPCI pump breaker left open.
RO-76-13	112726	3/24	4/6	C	WB	00	–	C	AY,BB	A	N	Component cooling water vault drain valve left open.

Table A2.7 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
RO-76-14	113280	3/23	4/19	C	SD	OO	-	C	AC,AW	D	C7	Leak rate of containment isolation valves exceed limits.
RO-76-17	113282 116877	3/24	4/23	C	RB	PP	-	C	AQ	D	N	Standby liquid isolation check valve leaks beyond Tech Spec limits.
RO-76-15	113281	3/24	4/23	C	SH	GG	-	C	AT	D	C7	Snubber inoperable due to low oil.
RO-76-16	112725	3/24	4/7	C	SF-C	Z	-	C	AV	B,C,E	C7, C4	Through wall cracks in HPCI pipe safe end.
RO-76-18	113283 116904	3/25	4/23 6/29	C	CH	PP	-	C	AW	G	N	Leak rates of two FW check valves exceeds limits.
RO-76-19	113284	3/27	4/26	C	RA	DD	-	C	AP,AO	B,E	S4	Jet pump restrainers cracked.
RO-76-20	113285	3/29	4/27	C	SF-C	PP	-	C	AR	D	C7	HPCI check valve leaked excessively.
RO-76-21	113286 116900	4/6	5/7 7/14	C	CJ	Z	-	C	AV	D	C7, c4	Through wall crack in isolation condenser safe end.
RO-76-22	113786	4/11	5/10	C	SF	F	-	B	AA	D	N	ECCS jocky pump breaker failed.
RO-76-23	113543	4/9	5/7	C	WB	JJ	-	C	AP,AV	D	N	Component cooling water pump vault leaks.
RO-76-24	113967	4/12	5/12	C	RB	OO	-	C	EH	D	C7	Set point drift in standby liquid isolation system relief valve.

Table A2.7 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
RO-76-25	114373	4/13	5/13	C	RB	I,J	-	B	OK	A	C3	Personnel were not evacuated during reload.
RO-76-26	113977	4/14	5/14	C	CJ	-	E	C	EH	D	N	Set point drift in IC pressure switch.
RO-76-27	113799 120671	4/13	5/13 12/15	C	RB	OO	-	C	EH	D	C7	Set point drift in a standby liquid control relief valve.
RO-76-28	113976	4/17	5/14	C	RB,IE	J	C	C	EG	D	N	CRD interlock failed.
RO-76-29	114171	4/21	5/20	C	RB,IE	S	C	C	EG	D	N	CRD permit interlock failed.
RO-76-30	114206	4/23	5/20	C	MB	P	-	C	AB	D	N	Excessive usage caused early failure of charcoal bed.
RO-76-31	115064	5/19	6/16	C	RB	OO	-	C	AU	A,G	C5	NaB_5 leaks into reactor vessel.
RO-76-32	114646	5/12	6/11	C	-	L	-	B	AM	B	N	Incorrect drive brakes on RB crane.
RO-76-33	115518	5/23	6/22	C	EE	N	-	C	AQ	D,G	C7	Swing DG failed to start.
RO-76-34	114645	5/25	7/7	B	SF,CE	OO	-	B	BB	D	S9	SRV failed to close.
RO-76-35	115517	5/27	6/24	B	SA	JJ	-	B	BT	A	N	Low N_2 pressure in the containment.

Table A2.7 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
RO-76-37	115733	5/27	7/6	B	BB	–	T,M	B	EH	D	N	Scram pressure switch actuated at upper limit.
RO-76-38	114644	5/26	6/9	D	MB	OO	–	B	AZ	G,H	C3	Chimney sample valve left open.
RO-76-39	115063	5/31	6/16	B	SF	–	–	C	OC	A	N	Failed to test LPCI and core spray on time.
RO-76-40	115732	6/7	7/2	B	MB	FF	–	B	BJ	D	C8	H_2 explosion in off gas system.
RO-76-41/ RO-76-42	115731 115730	6/16	6/28	B	RB	–	M,T	C	EH	D	N	High pressure scram switches set point above limit.
RO-76-43	115898	6/21	7/19	B	SF-C	–	I	C	EI	E	C1	HPCI level sensors damaged during construction.
RO-76-44	116271	6/21	7/16	B	SF-B	–	M,T	C	EH	D	N	LPCI pressure switch exceeds Tech Spec by 2%.
RO-76-47	115896	6/25	7/22	D	CF	F	T	B	AG	D	N	A LPCI pump failed to start.
RO-76-45	–	6/21	7/20	B	RB	–	M,T	C	EH	D	N	High pressure scram switch set point drift.
RO-76-48	116270	6/30	7/28	B	IA	–	B	B	EE	C	N	Average power range monitor gave incorrect reading.

Table A2.7 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
RO-76-49	116531	7/11	8/9	B	SF-D	OO	–	C	BA,AL	D	N	A core spray valve failed to open.
RO-76-50	116530	7/10	8/9	B	EE	F,N	–	B	BL	A	C7	DG cooling water pump breaker tripped due to excessive temperatures.
RO-76-51	117156	7/27	8/25	B	SH	DD,OO	–	C	BD	D	N	Diesel fire pump failed to start.
RO-76-52	117178	7/27	8/24	B	IB	–	M	C	EH	D	N	An ECCS drywell pressure switch drifted.
RO-76-53 RO-76-69	116891 120727	8/13 12/14	8/17 12/28	B	RC	–	–	–	BC	B	N	MAPLHGR limits not conservative and Dresden was notified by G.E.
RO-76-55	117970	8/17	9/15	B	MB	P	–	C	AQ	D	N	Flow rates less than required in standby gas treatment system.
RO-76-57	118186	8/21	9/20	D	IB	–	I	C	BD	D	N	Reactor water level switch set point drift.
RO-76-58	118187	6/10	9/23	B	SF-D	OO	–	C	AK	G	C7	A core spray valve failed to operate.
RO-76-59	118188	9/2	9/15	B	MB	OO	–	B	BC	B	C4	Logic error renders standby gas treatment system inoperable.
RO-76-60	118429	9/5	10/5	B	SF-C	Z	–	C	AX,AO	E	C1, C3	HPCI test line leaks due to bad well.
RO-76-62	119487	9/30	10/29	B	EE	N	R	C	EF	D	C7	Swing DG output erratic.
RO-76-61	–	9/15	10/14	B	SF-C	Z	–	B	AX	G	C3	Water from the HPCI test return line leaked into the sewer.

Table A2.7 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
RO-76-63	119513	10/3	11/1	B	MB	OO,P	-	B	AA	D	C7	Stack sample pump failed.
RO-76-64	120426	10/29	11/26	B	EE	N,HH	-	C	AC	G	C7	DG failed to start.
RO-76-67	120532	12/3	12/15	B	RB	RC	-	-	EI	B	N	Reanalysis shows upper limit of power for rod worth minimizers unconservative.
RO-76-66	-	11/13	11/23	D	SF-C	OO	-	C	AD	D	C8	HPCI injection valve had a severed valve stem.
RO-76-68	-	12/12	1/10/77	B	RB	J	-	C	CA	E	N	A CRD uncoupled while testing.
RO-76-70	-	12/18	1/13/77	B	EE	N	-	C	EC	H	N	During flushing of the turbine building oil drain, water leaked into the diesel's oil tank.
RO-76-71	-	12/23	1/20/77	B	-	-	-	-	OC	A	N	Several surveillance interval limits exceeded.
RO-76-72	-	12/28	1/26/77	D	RB	J	-	B	CA	E	N	A CRD uncoupled during startup.
RO-76-73	-	11/17	2/3/77	B	PC	BB	-	B	BT	D	N	The nitrogen storage tank level was low during the inerting process.

Table A2.7 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
RO-76-74	-	12/28	3/10/77	D	RB	J	-	B	OB	D	N	A control rod was withdrawn one notch and produced a larger worth than expected.

Table A2.8 Coding Sheet for Reportable Events for Dresden 2 — 1977

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
RO-77-01	122189	1/17	1/27	–	RC	–	–	–	OK	B	N	The MAPLHGR limits were reduced.
RO-77-02	122173	2/9	12/8	B	IA	–	N	B	EH	A	N	Set point drift in average power range monitor.
RO-77-03	130079	9/12	10/12	C	CB	00	–	B	BB	D	N	Recirc suction valve failed to open.
RO-77-04	122188	2/17	3/2	–	RC	–	–	–	OK	B	N	MAPLHGR adjusted to meet new ECCS models.
RO-77-05	122134	2/17	3/2	B	SF-B	00	T	C	EH	D	N	Set point drift caused LPCI to actuate.
RO-77-07	123039	2/23	3/24	B	IA	–	N,T	C	EH	D	N	Set point drift in intermediate range monitor.
RO-77-08	123038	2/25	3/24	B	CD,IB	–	M,T	C	EH	D	N	A main steam line's low pressure switch drifted.
RO-77-09	123088	3/16	3/21	B	CH	–	G	B	ED	G	N	Sudden introduction of a cold slug into feed-water.
RO-77-10	123796	3/20	3/31	D	SF-B	Z	–	C	AO	E	C7	Crack in LPCI discharge header.
RO-77-11	123799	3/22	4/4	B	EE	N	–	C	BC	D	C7	Swing DG failed to start on first attempt.

Table A2.8 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
RO-77-12	126203	3/21	4/19	B	MA	JJ	-	B	BU	B	N	Radwaste surge tank overflowed.
RO-77-13	124202	4/1	4/15	D	MA	Z	-	C	AW	D	C7	Excessive radwaste leakage from the drywell.
RO-77-14	125211	4/2	4/29	D	RB	J	-	B	CA	D	C7	CRD uncoupled.
RO-77-16	125038	4/2	5/2	D	CB	OO	T	B	BA	G	N	Recirculation isolation valve failed to close.
RO-77-17	125039	4/13	5/13	B	IF	OO	F	B	BA	D	N	Tip ball valve failed to open.
RO-77-18	125113	4/15	5/13	B	MB	-	N	B	EH	D	C7	Offgas monitoring system erratic due to instrument drift.
RO-77-19	124895	4/25	5/9	B	IA	N	-	B	BC	G	N	Two average power range monitors out of calibration.
RO-77-20	125592	5/29	6/28	B	CF	U	-	B	AX	D	C3	Unplanned release of radioactive liquid.
RO-77-21	126007	6/3	7/1	B	CF	OO	T	B	RC	D	N	RHR heat exchanger valve failed to open.
RO-77-22	126037	6/5	7/1	D	RB	J	-	B	CA	D	C7	CRD uncoupled.
RO-77-23	126613	6/5	7/5	D	IA	-	L	B	EH	D	N	Average power range monitor drifted.

Table A2.8 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
RO-77-24	127010	6/30	7/14	B	EE	N	-	C	BF	D	C8	Emergency power was lost to Dresden 3, but the swing diesel affects Unit 2. DG 2/3 tripped on overspeed.
RO-77-25	127926	7/12	8/11	B	EE	N	-	C	EI	C	C7	Emergency power jeopardized. DG 2/3 tripped on overspeed.
RO-77-26	127925	7/1	8/9	B	IA	-	M	C	EH	D	N	A reactor protection system pressure switch drifted.
RO-77-27	127977	7/9	8/4	D	RB	J	-	B	AB	D	S9	46 CR failed to fully insert.
RO-77-28	128224	7/28	8/26	B	WB	BB	-	C	AY	G,A	N	Component cooling water vault door left open.
RO-77-29	128443	8/2	9/1	B	RB	J	-	C	CA	G	C7	2 CRD's uncoupled.
RO-77-30	127976	8/2	8/16	B	SF-C	-	T	C	BA	G	C1	HPCI valve failed to open.
RO-77-31	128225	8/5	8/19	B	MA	JJ	-	B	BS	H	N	Damper and caustic day tanks overflowed.
RO-77-32	128891	8/15	9/7	B	SF-B	OO	-	C	BA	D	N	LPCI valve failed to open.
RO-77-35	129831	9/10	8/23	B	SF-C	OO,F	-	C	BF	D	C1	HPCI valve failed to open.
RO-77-36	129832	9/10	9/22	B	SF-C	NN	-	C	AL	D	C1	Speed governor failed on HPCI turbine.

Table A2.8 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
RO-77-38	130080	9/13	10/13	C	CB	OO	–	A	BB	D	N	MSIV failed to close at end of normal life.
RO-77-39	130012	9/8	10/7	B	CD	OO	T	B	AT,BL	D	N	Isolation signal received incorrectly from MSL temperature switch.
RO-77-39	130081	9/15	10/14	C	CF,SF, SA	FF	–	B	AW	D	N	Gasket failed on torus drain flange.
RO-77-41	130124	9/20	10/18	C	IA	–	L	C	EH	D	N	Average power range monitor drifted.
RO-77-42	130799	9/26	10/25	C	CA	KK	–	C	AO	E	C8	Jet pump clamp bolt keeper had broken welds.
RO-77-43	130914	9/29	10/28	C	IA	–	M	C	EH	D	N	Drywell pressure trip switch drifted.
RO-77-44	130917	9/30	10/28	C	SF-B	–	M	C	AC	D	N	A LPCI logic pressure switches failed.
RO-77-45	130082	10/4	10/18	C	CH	PP	–	C	AU	D	N	Leak rate through feedwater check valves greater than Tech Spec limit.
RO-77-46	130879	9/30	10/28	C	SH	GG	–	C	BT	D	C7	Two snubbers found with low oil.
RO-77-47	130083	10/1	10/14	C	RB	Y	–	C	RO	A,B	C7	CRD return nozzle cracked.
RO-77-42	130084	10/5	10/19	C	RB	Z	–	C	AR,AX	C	C7	Cracks in CRD and pipe safe end.

Table A2.8 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
RO-77-49	130880	10/7	10/31	C	SA	KK	-	C	AO	E	C8	Insufficient weld surface on 7 of 14 containers.
RO-77-50	130884	10/21	11/3	C	SA	FF	-	C	AQ,AW	D	N	Drywell air lock leaked.
RO-77-51	131704	10/30	11/29	C	EE	N,P	-	C	BE	A	C7	Swing DG tripped during test.
RO-77-54	131790	11/2	12/2	C	RB	J,I	-	C	CA	E	C7	CRD uncoupled.
RO-77-55	131789	11/3	12/2	C	EE	N	-	C	OJ	A	N	Swing DG declared inoperable.
RO-77-56	131761	11/4	12/2	C	SB	U	-	C	AY	D	C7	Containment cooling heat exchanger leaked.
RO-77-57	130996	11/1	11/15	C	RB	J	T	C	BZ	D	N	CRD select allowed two drives to be selected.
RO-77-58	130920	11/2	11/16	C	SA	OO	-	C	AW	E	C8	Three torus air sample valves leaked.
RO-77-59	131796	10/26	11/23	B	SF-C	OO,DD	-	C	OC	A	C7	HPCI surveillance test missed.
RO-77-60	131762	11/8	12/8	C	SA	OO	-	C	AO,AW	D	N	A torus suction line isolation valve leaked.
RO-77-61	131788	11/11	12/9	C	IB	-	I,T	C	EH	D	N	Reactor water level switch drifted.
RO-77-62	131787	11/9	12/9	C	CB	G	-	C	A	E	C4	Condenser pit pump cables misplaced.

Table A2.8 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
RO-77-65	132950	11/22	12/22	C	SF-B	OO	-	A	AW	G	N	Water spilled from LPCI valve during maintenance.
RO-77-66	132157	11/29	12/13	D	EE	N,DD	-	C	BD	B	C4	Swing DG cooling water pump failed.
RO-77-67	132951	11/24	12/23	C	SA	OO	-	C	BB,AW	D	C7	Drywell vent valve leaked.
RO-77-68	133081	11/29	12/29	C	IA	-	M,T	C	EH	D	N	Set point drift of pressure switch.
RO-77-69	133080	11/30	12/30	C	IA	-	H	C	BA	C	C8	Intermediate range monitor did not trip on loss of high voltage.
RO-77-71	132946	12/3	12/23	C	EE	N	-	C	BD	D	C7	Swing DG failed to start due to ruptured air starter diaphragm.
RO-77-72	133691	12/4	12/30	C	EE	N,F	-	C	BB	D	C8	DG failed to close onto breaker.
RO-77-73	133690	12/8	12/29	C	EE	F	-	C	BC	A	C8	LPCI breaker left in test position.
RO-77-74	133695	12/8	1/6/78	C	IA	-	H	C	EH	D	N	An intermediate range monitor drifted from set point.

Table A2.8 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
RO-77-75	132945	11/16	12/16	C	EE	N	-	C	BD	D	C7	Swing DG failed to start on first attempt.
RO-77-76	133044	12/29	12/29	B	SA	JJ	-	B	BT	A	N	N_2 storage tank level dropped.
RO-77-80	133617	12/16	12/30	B	SF-C	Q	-	C	AW	G	C1	HPCI steam exhaust line leaked.
RO-77-81	134074	12/22	1/5/78	B	IA	-	L	C	BC	A	C8	All average power range monitor flow bias setpoints maladjusted.
RO-77-82	133685	12/23	1/13/78	B	IA	-	T,M	C	EH	D	N	ECCS drywell high pressure switch setpoint drifted.
RO-77-77	132952	12/12	12/23	B	SA	OO	-	C	AG	G	N	Drywell vacuum breaker failed.
RO-77-70	-	12/02	12/15	C	EE	N	-	C	BD	B	S9	Loss of Emergency Power. Unit 2/3 DG out of service, Unit 2 DG failed to start.

Table A2.9 Coding Sheet for Reportable Events for Dresden 2 — 1978

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
RO-78-01	134489	1/3	2/2	B	EE	N	-	C	AL,BD	D	C7	DG failed to start.
RO-78-02	134250	1/5	1/18	B	RB	OO	-	B	AZ	G	C8	Standby liquid injection pump's suction valve inadvertently closed.
RO-78-03	134959	1/11	2/10	B	IF	F,OO	-	C	BB	D	N	Tip ball valve failed to close.
RO-78-04	142701	1/10	2/2	-	EC	-	-	B	BC	E	N	Two 125 V DC battery cables not routed conservatively.
RO-78-05	135766	1/27	2/21	B	IF	-	I	B	EH	D	N	Instrument level drift.
RO-78-06	136348	1/30	3/1	B	SC	-	I	C	EI	G	N	O_2 analysis mis-calibrated.
RO-78-07	135767	2/17	2/21	B	MB	-	-	C	OK	H	N	Test run for insufficient time.
RO-78-08	135893	2/10	2/24	D	SD	OO	-	C	AG	G	C7	Excess grease in shaft seals caused vacuum breaker binding.
RO-78-09	135894	2/12	2/24	B	CG	P	-	B	BU	H	N	Coolant conductivity exceeded limit for two hours.
RO-78-11	136347	2/16	3/2	B	IE	-	I	C	EH	D	N	A reactor switch drifted.

Table A2.9 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
RO-78-12	136465	2/18	3/17	B	CD	00	U	C	AL	G	N	A MSIV's pilot valve exceeded temperature limit.
RO-78-15	135972	2/25	3/9	D	SD	00,HH	-	C	AW,AY	D	N	A continuous air monitor valve failed to close.
RO-78-16	136381	2/24	3/17	B	SD	00	-	C	AD	D	C7	Reactor building to torus vacuum breaker failed open.
RO-78-18	142567	2/27	3/29	B	SA	JJ	-	B	BS	A	N	N_2 level dropped low in storage tank.
RO-78-20	142704	3/7	6/30	B	EE	N,HH	-	C	BD	D	C7	Swing DG failed to start.
RO-78-21	137336	3/8	4/4	B	EE	N	-	C	EI	D	C7	Swing DG failed to run due to governor mis-calibration.
RO-78-23	137250	3/20	4/3	B	SF-C	00	-	C	AW	B	C1	HPCI control valve leaked.
RO-78-24	137335	3/21	4/4	B	CJ	-	-	B	OJ	H	S7	Isolation condenser rendered inoperable while HPCI unavailable.
RO-78-25	137334	3/27	4/7	B	BA,IA	-	N	B	EI	A	C8	Typo in main steam line radiation monitor tech specs.
RO-78-26	137830	3/30	4/21	B	SF-C	00	-	A	OA	D	C1	HPCI taken out of service for maintenance.

Table A2.9 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
RO-78-27	138923	4/18	5/18	B	MA	JJ	-	B	OK	A,H	C7	Activity level exceeds limit in storage tank.
RO-78-28	137855	4/21	5/5	B	SF-B	-	C	B	OK	B	C4	LPCI/LOCA logic error.
RO-78-32	140559	5/1	6/9	B	SD	OO	-	C	AG	D	C7	Two torus drywell vacuum breakers inoperable.
RO-78-33	139748	5/22	6/21	C	EE	N	-	C	BC	D	C7	DG tripped on overspeed.
RO-78-34	139583	5/28	6/22	B	CD	-	U,T	B	EH	D	N	A main steam line isolation switch tripped prematurely.
RO-78-35	140582	5/30	6/9	B	FC	KK	-	C	AV,AO	D	C8	Cracks found in spent fuel storage racks.
RO-78-36	139877	6/5	7/5	B	MB	S,Z	-	B	AW	D	C7	Train A of standby gas treatment plant failed on two occasions.
RO-78-38	140504	6/19	6/30	B	CD	-	U,T	B	EH	D	N	A main steam line isolation switch tripped prematurely.
RO-78-40	140047	6/26	7/28	B	IA	-	L	C	EI	D	N	Set point drift in average power range monitor.
RO-78-41	140045	6/30	7/26	B	EE	N	U,T	C	EB,EI	G	C7	Miscalibration of thermal overload switch caused swing DG to trip.

Table A2.9 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
RO-78-42	140065	7/8	8/7	B	HA	OO,HH	-	C	AC	D	C8	Turbine control valve failed to "fast close."
RO-78-43	140512	7/28	8/25	B	IA	I	P,R	B	EG	D	C8	A rod worth minimizer circuit failed during startup.
RO-78-44	140514	8/1	9/25	D	IA	-	R	C	EE	D	C8	Average power range monitor set point out of limits.
RO-78-45	140517	8/7	8/25	D	SF-A	-	T	C	AR	D	C1	ADS auto-blowdown permissive switch set point drift.
RO-78-46	140153	8/7	9/1	D	CB	Z	-	C	AV	E	S9	Two cracked welds in recirc pump suction bowl drain line.
RO-78-47	140230	8/8	9/8	D	RB	-	-	C	OI	G,H	N	Operator misread boron concentration.
RO-78-50	140162	8/24	9/7	B	EE	N	-	C	BD	D	S9	DG failed to start on first attempt with swing DG unavailable.
RO-78-52	141480	9/22	10/19	D	EE	N	-	C	BD	D	C7	Swing DG failed to start on first attempt.
RO-78-53	141856	9/26	10/17	B	IE,IA	OO	E,T	C	AZ	G	N	One of 16 main steam line flow switches valved out.

Table A2.9 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
RO-78-54	141459	9/30	10/24	B	SD	OO,HH	-	B	AY,ED	D	C7	Redundant vacuum breaker failed open.
RO-78-55	141203	9/30	10/13	B	SF-C	-	M,T	C	EH	D	C1	HPCI tripped on low suction pressure.
RO-78-56	141724	10/12	11/1	B	SD	OO	M,T	C	AZ	A,G	N	DP root valve left closed.
RO-78-57	141391	10/13	11/9	B	SF-D	OO	-	C	BA,AK	G	N	A core spray test valve failed to open.
RO-78-58	141392	10/18	11/13	B	SF-B	OO	-	C	BA,AK	G	N	LPCI suction valve failed to close.
RO-78-61	152326	11/8	12/8	B	SD	OO	-	B	BB	D	N	One of 5 containment vent valves failed to close.
RO-78-62	142327	11/19	12/8	B	SH-A	S,HH	-	B	ED	D	N	Containment purge valve fuse blown.
RO-78-63	142648	11/23	12/19	B	CD	OO	-	C	BI	D	N	Two MSIV closure times out of limits.
RO-78-64	142688	11/24	12/21	B	MA	JJ	-	C	AR	B	N	Leak found in waste collection tank.
RO-78-66	146495	12/16	1/9/79	B	EE	N	-	C	BD	D	C7	Swing diesel failed to start.
RO-78-67	146496	12/17	1/16/79	B	BB	OO	N	B	BB	D	N	Continuous air monitor isolation valve failed to close.

Table A2.9 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
RO-78-68	146594	12/20	1/8/79	B	CB	OO,F	-	B	BB,AF	D	N	"A" recirculation bypass valve failed to open.
RO-78-69	146497	12/10	1/9/79	B	BB	-	-	-	OC	A	N	Offgas sample not taken.

Table A2.10 Coding Sheet for Reportable Events for Dresden 2 — 1979

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
LER79-01	147238	1/3	1/31	B	SB	JJ	-	B	BT,OK	A	N	Torus water level above limit.
LER79-02	149862	1/3	6/5	B	SF-C	QQ	-	C	BD,BC	D	C1	HPCI failed to start.
LER79-03	147249	1/5	1/31	B	SF-D	-	M,T	C	EH	D	N	Setpoint drift in core spray pressure switch.
LER79-04	147352	1/11	1/24	B	SF-B	OO	-	C	BB	D	N	LPCI flow test valve failed to close.
LER79-05	147353	1/19	2/9	B	CD	-	N	B	EE	D	N	A main steam line radiation monitor failed.
LER79-08	147354	1/26	2/9	B	SF-B	OO	-	C	BB	D	N	A LPCI low flow discharge valve failed to close completely.
LER79-09	147355	1/21	2/3	B	BB	OO,P	-	C	BA,AG	E	N	Plant chimney installed incorrectly and plugged.
LER79-10	147356 150868	2/2	2/10	D	FD	X	-	B	AF	B	C3	Refueling panel blow out panel blew out.
LER79-13	148322	2/23	3/23	B	EE	N,DD,F	-	B	BC	D	C7	Swing DG failed to run.
LER79-14	149358	3/5	3/30	B	EE	N	-	C	BD	D	C7	Swing DG failed to start on first attempt.
LER79-15	149357	3/21	4/4	C	RB	Z	-	C	AR	D	C8	GE warns of possible cracks in control rod blade tubing.
LER79-16	149356	3/26	4/6	C	SF-B	PP	-	C	AO	E	N	LPCI check valve leaked excessively.

Table A2.10 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
LER79-17	149355	3/27	4/9	C	SD	OO	–	C	AW	D	C7	Six containment isolation valves exceed leak rate limit.
LER79-19	149353	3/22	4/19	C	CH	Z	–	C	AV	B	C7	Cracks in reactor feed pump minimum flow line.
LER79-20	149701	4/13	5/4	C	SD	FF	–	C	AW	G	N	Faulty handwheel packing resulted in leaky airlock.
LER79-21	149702	4/15	5/14	C	SF-B	DD	–	C	BD	D	C8	LPCI pump 2"A" failed to run.
LER79-22	149354	3/31	4/30	C	EE	N	–	C	BC	D	C7	DG failed to reach rated speed while the swing DG was unavailable.
LER79-23	149352	4/6	4/26	C	CA	Z	–	C	AO	E	C8	Bad weld on reactor head vent line.
LER79-24	149571	4/24	5/24	C	EE	N	–	C	BD	G	C7	Air lines to starter motor reversed.
LER79-25	149572	4/25	5/24	C	IF,HA	–	M	C	EI	D	N	Turbine pressure switches set point drift.
LER79-26	149570	4/28	5/24	C	CB	F	–	C	BA	D	N	A reactor protection system breaker did not trip.
LER79-27	149568	5/2	5/30	B	IA	–	M	B	AU,EI	G	N	Miscalibrated yarways caused turbine trip.
LER79-28	149567	5/8	5/25	B	SE	Z	O	B	BN,EG	D	N	N_2 inerting valve damaged.

Table A2.10 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
LER79-29	14727	5/8	6/5	B	IB	-	MT	C	BA	D	N	One of 4 ADS permissive pressure switches failed.
LER79-31	149728	5/9	6/7	B	HH	Z	-	C	AV,AW	D	C8	Crack in feed system piping weld.
LER79-32	149569	5/1	5/30	D	IA	G	H	B	AL	G	N	Two intermediate power range monitors failed downscale.
LER79-33	149565 153316	4/28	5/25	C	SD	OO	T	C	CA	D	N	TIP fails to retract during refueling.
LER79-34	150051	5/24	6/12	B	EE	N	-	C	BD	D	C7	The Unit 2/3 swing diesel failed to start due to failed lower air starter motor.
LER79-34	150051	5/30	6/12	B	EE	N	-	C	BD	D	S9	After declaring Unit 2 DG inoperable, Unit 2/3 DG was tested and failed to start.
LER79-37	150048	5/30	6/22	B	EE	N,HH	-	B	BC	H	C7	Unit 2 DG inoperable due to operator error.
LER79-39	150050	5/28	6/15	B	CA	-	T	B	AY	H	N	Operator error caused loss of recirc MG set control power.
LER79-40	150361	6/5	7/3	B	MA	Z	-	B	AW	D	C3	Radwaste reboiler leaked.
LER79-41	150049	6/6	6/15	B	CA	LL	-	B	AA	D	N	Transformer failure caused recirc pump trip.

Table A2.10 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
LER79-42	150360	6/8	7/2	B	SF-C	-	E,T	C	EH	D	C1	Setpoint drift in HPCI high steam flow switch.
LER79-43	151174	6/21	7/19	B	MA	JJ	-	B	BU	H	N	Radwaste level in sample tank exceeded limit.
LER79-44	151173	7/9	8/3	B	EE	N,DD	-	C	BD	D	C7	Swing diesel generator tripped.
LER79-45	151175	7/24	8/21	B	EE	N,FF	-	C	BT	D	C7	DG cooling water flange O-ring leaked.
LER79-46	151257	7/29	8/28	B	SE	JJ	-	B	BT	H	N	N_2 storage tank level exceeded limit.
LER79-47	152331	8/16	9/11	B	EE	N,F	-	B	OA	D	N	DG circuit breakers tripped open.
LER79-48	152330	9/8	9/26	B	BB	DD	-	B	AA	D	N	Chimney sample pump failed.
LER79-49	152622	9/18	10/10	D	EE	N	-	-	OC	H	N	Fuel oil samples not taken in April.
LER79-50	152621	9/19	10/12	B	AD	L	-	C	OC	H	N	Overhead crane pre-use test not adequately performed.
LER79-51	152619	9/25	10/23	B	IA	-	M,T	C	EH	D	N	One of 4 drywell pressure switches drifted.
LER79-52	152620	10/1	10/18	B	EE	N	P	B	ED	G	N	DG auto start relay shorted out by water.
LER79-53	153578	10/3	11/1	B	MA	-	E,T	B	AQ	D	N	"A" standby gas treatment plant trips on initial signal.

Table A2.10 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
LER79-54	153577	10/4	10/29	B	SD	KK	-	C	AD	D	N	Containment vent line pipe supports inoperable.
LER79-55	153516	10/4	10/30	B	SD	OO	-	C	AQ	D	N	Drywell torus vent valves failed to close.
LER79-56	152618	10/4	10/23	B	CJ	OO,S	-	B	EE	G	N	A short circuit caused a relief valve to become inoperable.
LER79-57	153575	10/19	10/31	D	IA	-	M,T	C	EH	C	N	Three MSIV scram limit switches out of limits.
LER79-58	153574	10/20	11/13	B	CJ	-	E,T	C	EH	D	N	Isolation condenser high flow switches drifted.
LER79-59	153573	10/20	10/20	B	SF-D	KK	-	B	AE	E	N	Pipe support bolts found defective.
LER79-60	153572	10/30	11/28	B	AB	OO	-	C	OK	A	N	Fire protection valves not cycled as required.
LER79-61	153571	11/9	12/6	B	AB	JJ,OO	-	B	EH	D	N	Setpoint drift in cardox (fire protection) storage tank.
LER79-66	153495	12/13	1/11/80	B	MA	JJ	-	B	OK	A	C7	Radwaste activity exceeded limit.
LER79-62	153570	11/10	12/6	B	MB	S	-	B	EE	B	N	Fuse in standby gas treatment plant blew.
LER79-63	153569	11/14	11/30	D	IA	-	N	C	EH	D	N	Setpoint drift in source range monitor.

Table A2.10 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
LER79-64	153748	11/24	12/21	B	SD	OO	-	B	AG	D	N	Torus to drywell vacuum breaker failed.
LER79-65	154820	11/30	12/29	B	MA	OO	-	B	BA	D	N	Standby gas treatment valves failed to open.
LER79-67	153494	12/18	1/7/80	B	EE	N	-	C	BD	D	C7	Swing DG failed to start.
LER79-68	153940	12/29	1/23/80	B	AB	-	-	C	OE	E	N	Fire stop integrity inadequate.

Table A2.11 Coding Sheet for Reportable Events for Dresden 2 — 1980

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
LER80-01	154620	1/6	1/30	B	IA	–	T	C	BA	D	N	Turbine control valve closure scram failed to reset.
LER80-02	154618	1/4	1/30	B	CD	–	M,T	C	OC	A	N	Main steam line low pressure switch test not performed.
LER80-03	155197	1/14	2/13	B	IA	–	I,T	C	EH	D	N	Reactor water level sensors drifted.
LER80-04	154616	1/18	1/30	B	CD	OO	–	C	AA	D	N	MSIV closure times out of limits.
LER80-06	154927	1/19	2/15	B	MA	JJ	–	B	BU	H	N	Waste tank activity exceeded limit.
LER80-07	155295	1/25	2/19	B	SE	–	O	C	EH	D	N	Drywell O_2 monitor drifted.
LER80-08	155312	1/26	2/22	B	EE	N,U	–	B	AU	D	C7	Swing DG heat exchanger leaked.
LER80-11	160002	3/18	4/3	B	CJ	–	E,T	B	EH	D	N	Isolation condenser flow switches drifted.
LER80-14	157163	4/29	5/27	B	IA	–	P	B	EG	D	N	A half scram was initiated by a main steam line relay failure.
LER80-15	158556	5/2	5/27	B	SA	OO	–	C	AB	D	C7	Torus to drywell vacuum breaker failed.
LER80-16	158554 163598	5/15	5/20 10/8	D	SF-B	Z	–	C	AW	B	N	Cracks in 3/4" LPCI drain line.

Table A2.11 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
LER80-17	157698	5/12	6/5	D	SF-C	OO	-	B	BB	D	C1	HPCI isolation valve failed to closed.
LER80-18	157686	5/12	6/3	B	SF-C	OO	-	B	BB	G	C1	HPCI isolation valve failed to close.
LER80-19	156216	2/29	3/21	B	AA	R	-	B	EE	D	N	Control room emergency fan failed.
LER80-19	160264	5/28	6/24	B	IB	-	N,C	B	EG	D	N	Half scram initiated by faulty main steam line radiation monitor.
LER80-20	160266	6/5	7/2	B	SD	OO	T	C	AL	D	N	Torus to drywell vacuum breaker failed to give open signal.
LER80-21	158247	6/7	7/2	C	SA	-	M	C	BK	H	N	Drywell to torus DP dropped below limit.
LER80-22	158692	6/11	6/24	B	RB	KK	-	B	AH	B	N	CRD scram discharge volume did not meet seismic specifications.
LER80-23	158704	2/28	7/14	B	SA	OO	-	B	BA	D	N	Two torus vent valves failed to open.
LER80-24	159531	7/21	8/14	B	SE	-	U,T	B	BL	D	N	Drywell oxygen concentration indicator failed to downscale.
LER80-25	159243	7/21	8/8	B	CJ	-	E,T	A	EH	D	N	Setpoint drift in isolation condenser condensate flow switch.

Table A2.11 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
LER80-26	159527	7/25	8/15	B	SF	OO	-	C	AZ	G	N	One of 4 drywell pressure switches valved out.
LER80-27	159532	7/25	8/19	B	EE	U	-	B	AT	D	C7	DG heat exchanger leaked.
LER80-28	159447	7/27	8/14	B	RB	I,J	-	B	CA	D	N	A CR became uncoupled.
LER80-29	159439	7/28	8/18	D	CG	Z	-	C	AO	E	N	Reactor water cleanup system instrument line leaked.
LER80-30	160208	8/18	9/15	B	SF-B	DD	-	B	AQ,AU	D	N	LPCI pump seal leaked.
LER80-31	160362	8/27	9/19	B	AB	N	U	C	AC	A	N	Diesel fire pump out of service too long.
LER80-32	160361	8/29	9/25	B	AB	N	-	B	OC,OK	A	N	Diesel fire pump day tank samples not taken.
LER80-33	160087	9/4	10/1	B	CF,SF-B	U	-	B	AR	D	N	LPCI/RHR heat exchanger leaked.
LER80-34	160461	9/8	9/26	B	AB	-	-	-	OK	A	N	Record of 1979 fire protection system flush lost.
LER80-35	160824	9/11	10/22	D	RB	I,J	-	C	CA	D	C7	A CR uncoupled.
LER80-36	160823	9/11	10/7	B	AB	-	U	C	OC	H	N	Heat detector surveillance not performed.
LER80-37	160056	9/15	10/8	B	RB	I,J	-	B	CA	D	N	A CRD uncoupled.

Table A2.11 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
LER80-38	160925	10/11	10/22	D	RB	I,J	-	C	CA	D	C7	A CR uncoupled.
LER80-39	160566	10/11	10/16	B	SF-C	00	T	C	EE	D	C1	HPCI steam supply valve failed to open.
LER80-41	160524	12/12	10/16	B	BA	-	N	C	EH	D	N	Area radiation monitor drifted.
LER80-42	161793	11/2	11/12	D	RB	JJ	-	C	OC	A	N	Ultrasonic test of scram discharge volume missed.
LER80-43	161871	11/13	12/3	B	HC	-	M,T	C	EH	D	N	Minor drift in condenser pressure switch.
LER80-44	161903	11/24	12/12	B	CI	00	A	B	OA	D	N	A relief valve was inoperable in the safety mode.
LER80-45	163379	12/1	12/30	B	IB	-	M,T	C	EH	D	N	LPCI high drywell pressure switches drift.
LER80-46	163371	12/2	12/29	D	RB	JJ	I,A	B	EG	E	S9	SDV high level alarm failed.
LER80-47	163515	12/4	12/27	D	IB	-	M,T	C	EH	D	N	MSL high pressure switches drift.
LER80-48	163360	12/19	1/14/81	B	SF	00	-	B	AZ	G	C1	ADS valve found closed.
LER80-49	163538	12/29	1/23/81	B	SD	00	-	C	AQ	D	N	Reactor building vent isolation valve failed to open.
LER80-50	163537	12/31	1/23/81	B	SH-B	-	M,T	C	EH	G	N	Containment spray pressure switch set point was too high.

Table A2.12 Coding Sheet for Reportable Events for Dresden 2 — 1981

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
LER81-01	163646	1/4	1/28	C	RB	-	I,A	B	EG	E	C8	Three of four scram discharge volume high level alarms failed.
LER81-02	164596	1/5	2/2	C	SF-B	OO	-	C	BB	D	N	LPCI test spray valve failed to close.
LER81-04	-	1/2	1/28	B	IB	-	M	B	OC	H	N	Electro-hydraulic effect low oil pressure scram test not performed.
LER81-05	164228	1/17	2/5	C	SD	OO	-	C	AW	D	N	Containment isolation vent valve leaked.
LER81-06	164594 167618	1/23	2/4	C	CG	PP	-	C	AW	D	C8	Feedwater check valve leaked.
LER81-07	164275	1/22	2/17	C	CI	-	U,T	C	AW	D	N	Steam line area temperature switch failed to trip.
LER81-08	164231	1/23	2/20	C	CD	OO	-	C	AW	D	N	Main steam isolation valve leaked.
LER81-09	164232	1/29	2/26	C	EB	F	-	C	AQ,BA	D	N	Reactor protection system bus breaker failed to open.
LER81-10	164573	2/18	3/9	C	CB	GG	-	C	AW	D	N	Five snubbers failed bench test.
LER81-11	166039	2/26	4/27	C	MA	OO	-	B	OJ,BC	H	C3	Unsampled water was discharged to the river.

Table A2.12 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
LER81-13	164684	3/2	3/13	C	SF-C	Z,GG	-	B	OA	B	N	HPCI pipe supports require modifications to meet seismic qualifications.
LER81-14	165479	3/4	3/31	C	MA	-	-	-	OJ	H	N	Radwaste discharge tests calculated incorrectly.
LER81-15	165882	3/19	4/14	C	AB	-	U	C	OC	G	N	Heat detectors not tested in time.
LER81-16	167916	4/8	7/13	C	MA	-	-	-	OK	A	N	Liquid effluent monitor test not performed on time.
LER81-17	171113	4/8	5/1	C	ED	C	-	B	OK	A	N	Battery cell temperature readings were not performed.
LER81-18	166087	4/26	5/4	C	SH-B	PP,Z	-	C	AW	B	N	LPCI check valve drain line cracked.
LER81-19	166034	4/26	5/1	C	CE	Z	-	B	AO,AW	E	N	Isolation condenser line socket weld cracked.
LER81-20	166088	4/26	5/5	C	CJ	Z	-	C	AW	D	N	Two cracks found in reactor head spray line.
LER81-21	-	4/27	5/21	C	SD	OO	-	C	BB	D	N	Primary containment isolation valve failed to close.
LER81-22	166074	5/3	5/14	C	CB	Z	-	C	AW,AO	E	N	Coolant recirculation leakoff line cracked.
LER81-23	166462	5/11	6/3	B	SA	-	-	C	OJ	H	C8	Torus water volume exceeded limit.

Table A2.12 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
LER81-24	166610	5/13	6/5	B	PC	-	-	-	OC	H	N	Standby liquid control tank not sampled in time.
LER81-25	166545	5/18	6/10	B	CE	OO	-	B	BC	H	N	Isolation condenser level indication valved out.
LER81-26	166537	5/14	6/12	B	IE	G	M	B	AL,EF	D	N	Safety relief valve acoustic monitor failed.
LER81-27	166382	5/21	5/28	B	EE	DD,G	-	C	ED	D	N	Diesel generator oil pump failed.
LER81-28	166563	5/20	6/10	B	SF-D	OO	-	C	BA	D	C6	Core spray valve failed to open.
LER81-29	166455	5/23	6/18	B	CB	DD	-	B	AQ,ED	D	N	Recirculation pump motor tripped.
LER81-30	166634	5/26	6/8	B	SF-D	Z,OO	-	B	BC,OA	G	C6	HPCI steam line filled with water due to a reversed valve.
LER81-31	166554	5/27	6/10	B	RB	OO	-	B	BC,OJ	H	N	Scram accumulator charging water valve was closed.
LER81-32	166548	6/1	6/9	B	RB	J,OO	-	C	BC,AG	D	N	Control rod drive exceeded insertion limit.
LER81-33	166566 172491	6/2	6/11	B	SF-C	X	-	B	AL	D	C8	HPCI oil pump failed due to loose stator winding.

Table A2.12 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
LER81-34	166562	6/2	6/10	B	RB	I,J	-	C	AS	D	N	Control rod separated from drive.
LER81-35	-	6/1	6/26	B	MB	Z	-	B	AL	D	N	Stack gas filter assembly sample line uncoupled.
LER81-36	166546	6/2	6/9	B	RB	J,OO	-	C	BC,AG	D	N	Control rod drive exceeded insertion limit.
LER81-37	166716	6/7	7/2	B	SA	-	K	C	EF	D	N	Vacuum breaker position indication was lost.
LER81-38	166714	6/9	7/8	B	SA	F	-	C	AG	D	N	Containment vacuum breaker was found binding.
LER81-39	167167	6/24	7/21	B	IA	-	M,T	C	EH	D	N	Turbine pressure switch set point drifted.
LER81-40	167169	6/24	7/21	B	IA	-	P	C	BA	D	N	Reactor protection system relay failed to open.
LER81-41	167170	6/24	7/22	B	IB	-	I	C	AG	D	N	Reactor water level instrument failed.
LER81-43	167928	7/2	7/28	B	IA	-	M,T	B	EG	D	N	Pressure switch in protection system failed.
LER81-44	167927	7/3	7/31	B	IA	-	L,P	B	EG	D	N	Power range monitor failed to initiate a rod block.

Table A2.12 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
LER81-45	168037	7/6	8/3	B	MC	G	-	B	EE	D	N	Reactor building radiation monitor failed.
LER81-46	168133	7/9	8/6	B	MC	-	N,R	ED	B	D	N	Reactor building vent radiation monitor failed.
LER81-47	168109	7/15	8/13	B	CB	-	-	-	OC	H	N	Reactor coolant samples not taken prior to startup.
LER81-48	168520	7/24	8/20	B	IA	-	M,T	C	EH	D	N	Reactor pressure system turbine pressure switch set point drift.
LER81-49	168482	8/14	8/24	B	CD	OO	-	C	BC	D	N	MSIV closure time too fast.
LER81-50	168839	8/15	9/11	D	SD	OO,HH	-	C	BB	D	N	Containment isolation valves failed to close.
LER81-52	169079	8/23	9/18	B	IX	-	M,T	B	AA	D	N	Pressure switch failure caused generator load reject scram to be bypassed.
LER81-53	168619	8/17	9/3	B	BA	-	N	C	EH	D	N	Refueling flow radiation monitor set point drift.
LER81-54	169238	8/27	9/23	B	IA	-	M,T	C	EH	D	N	Reactor protection system pressure switch set point drift.

Table A2.12 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
LER81-55	172279	8/30	9/23	B	MA	Z	-	B	AR,AW	D	N	A radwaste system pipe corroded and leaked.
LER81-56	169357	8/28	9/21	B	SA	-	M,T	C	AA	D	N	High drywell pressure switch set point incorrect due to switch wear.
LER81-57	168894	9/3	9/11	B	SF-C	Z,GG	-	B	HD	D	N	HPCI inoperable due to broken hangers.
LER81-58	169211	9/6	10/5	B	MB	S	-	B	BG	D	N	Standby gas treatment train B tripped when a fuse blew.
LER81-60	169349 171115	9/13	10/13	B	MB	-	-	-	OL,OJ	H	N	Operator did not retake off-gas samples as required.
LER81-61	170193 172516	10/5	10/22	B	SF	GG	-	B	HH	D	N	A pipe hanger in the component cooling water system broke.
LER81-62	170033	10/11	11/9	B	SA	-	I,A	B	EH	D	N	Torus water level alarm setpoint drift.
LER81-63	170032	10/17	11/13	B	RB	J,OO	-	B	AG	D	N	CRD failed to scram due to sticking scram solenoid valve.
LER81-64	170140	10/29	11/23	B	SA	-	M,T	C	EH	D	N	Vacuum breaker pressure switch setpoint drift.
LER81-65	171024	11/3	11/16	B	CE	OO	-	C	BI	D	C7	Three MSIV closure times exceeded limits.

Table A2.12 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
LER81-66	171117	11/9	12/7	B	CH	Z	-	B	BC,AO	D	N	The feedwater heater emergency spill line had a cracked weld.
LER81-68	171049	10/21	11/18	B	ED	C	-	B	OC	H	N	Storage battery surveillance test performed late.
LER81-69	171119	11/19	12/3	B	SF-D	QQ,Z	-	B	BC,OA	D	N	A core spray valve operator fell off its mounting.
LER81-70	171122	11/20	12/2	B	SF-D	QQ,Z	-	B	AV,AO	D	N	Core spray valve operator flange weld cracked.
LER81-71	171793	11/26	12/23	B	SD	S	-	B	BG	D	N	Control power lost to primary containment isolation valves.
LER81-72	172419	12/3	12/31	B	MB	Z	-	B	AQ	D	N	Standby gas treatment system had a plugged orifice.
LER81-73	172277	12/14	1/13/82	B	CJ	-	M,T	B	BC	G	N	Isolation condenser pressure switch valved out.
LER81-74	171576	12/14	1/13/82	B	IA	OO	M	B	BC	G	N	Low pressure instrument valve left closed.
LER81-75	172266	12/15	1/11/82	B	MC	-	-	C	OC	H	N	Main coolant sample not taken in time limit.
LER81-76	172260	12/16	1/15/82	B	RB	J	-	B	EH	D	N	Setpoint drift in control rod block channel.

Table A2.12 (Continued)

Number	NSIC Accession Number	Event Date	Report Date	Plant Status	System	Equipment	Instrument	Component Status	Abnormal Condition	Cause	Significance Category	Comment
LER81-77	172259	12/16	1/15/82	B	IA	–	M,T	B	BC	G	N	ATWS pressure switches valved out of service.
LER81-78	171575 172107	12/23	1/27/82	B	SF-A	00	–	C	BA	D	S7	ADS valve failed while HPCI was declared in-operable.
LER81-79	172255	12/23	1/14/82	B	SF-C,AB	–	G	B	OA	D	C8	Fire protection system actuation rendered HPCI inoperable.
LER81-80	172754	12/28	1/25/82	B	IA	–	P	C	AG	D	N	Time delay relay in reactor protection system failed.

Appendix B

CALCULATION FOR FAILURE RATES ON DEMAND

B.1 Emergency Power System

Estimate of System Demands

The number of demands is estimated from the test frequency and the number of transients that initiate emergency power. Monthly testing is specified in the Technical Specifications, however, weekly testing was instituted on December 2, 1977 to provide additional information about diesel generator performance. Therefore, the number of demands is estimated to be:

$$2 \text{ diesel generators} \times 12 \frac{\text{tests}}{\text{year}} \times 7.9 \text{ years} + 52 \frac{\text{test}}{\text{year}} \times 4.1 \text{ year} = 564 \text{ demands.}$$

Single Diesel Failure Rate Upon Demand

Forty-four failures upon demand were identified, thus the failure rate upon demand becomes

$$\text{failure rate upon demand} = \frac{44 \text{ failures}}{564 \text{ demand}} = 0.078 \ .$$

This is a factor of two to three greater than median number reported by WASH-1400 but is within the range reported by WASH-1400 of 0.01 to 0.1 per demand.

Failure of Emergency Power

Two emergency power failures were identified for Dresden 2 and these events are discussed in the text in Sect. 4.5.2.10. The event occurring on June 12, 1979 (RO 79-34) is considered a failure upon demand. The unit 2 diesel failed to start, after the unit 2/3 diesel had failed a test.

The event on December 2, 1977, ocurred 72 h after the unit 2/3 diesel failed during a test due to problems with the cooling water pump. This event will be considered as unabailability of the emergency power system. The number of demands is taken as 1/2 the number used in the single diesel calculations, since the emergency power system is tested when both diesels are tested. Thus the unavailability can be estimated as the failure rate upon demand plus the unavailability due to system failures over 12 years (105,120 h), or

$$\text{Emergency Power Unavailability} = \frac{1 \text{ failure}}{282 \text{ demands}} + \frac{72 \text{ h}}{105,120 \text{ h}} = 0.0042 \ .$$

This is about five times as high as the WASH-1400 median for emergency power of a two unit plant utilizing four diesel generators. It is, however, within the WASH-1400 range of 0.0001 to 0.01. The 0.0042 agrees quite well, however, with the emergency power failure rate of 0.005. This estimate was derived from historical data for all BWRs between ten years (1969–1979), as part of the accident sequence precursor project.[16]

High Pressure Coolant Injection

Estimate of System Demands

Different tests are performed on HPCI at different test schedules; however, valves are tested at least monthly, thus a monthly test schedule will be assumed. The number of demands is estimated from the test frequency, the number of transients that initiate HPCI, and the number of failures since HPCI must be successfully tested prior to being declared operable. Therefore, the number of demands becomes:

$$12 \frac{\text{tests}}{\text{years}} \times 12 \text{ years} + 1 \begin{array}{c}\text{Loss of}\\ \text{feedwater}\end{array} + 15 \begin{array}{c}\text{tests after}\\ \text{repair}\end{array} = 160 \text{ demands} .$$

The estimate of HPCI failure upon demand becomes:

$$\text{failure upon demand} = \frac{15 \text{ failures}}{160 \text{ demands}} = 0.094 .$$

The WASH-1400 estimate for single failures which best represents a failure upon demand is 0.013. The Dresden estimate is roughly seven times this estimate. The HPCI failure rate upon demand computed in the accident sequence precursor (ASP) program is 0.05, about 1/2 the Dresden 2 estimate.[16] No bounds have been computed for the ASP estimate or the Dresden 2 SEP estimate.

Conclusions

It appears that the availability of the Dresden 2 electrical power system is what would be expected or better based upon historical data, but not as good as the WASH-1400 estimate. The High Pressure Coolant Injection System, however, is roughly a factor of two greater than what is estimated using historical data and seven times the WASH-1400 estimate. It should be noted, however, these estimates are sensitive to the assumed number of demands.

APPENDIX G

NRC STAFF CONTRIBUTORS AND CONSULTANTS

This Safety Evaluation Report is a product of the NRC staff and its consultants. The NRC staff members listed below were principal contributors to this report. A list of consultants follows the list of staff members.

NRC Staff

Name	Title	Branch
A. Chu	Nuclear Chemical Engineer	Accident Evaluation
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Consultants

Name	Company	Topic	Report Date
F. Farmer	EG&G, Idaho	III-10.A	December 1980
		VI-7.C.1	September 1979
		VIII-3.B	December 1980
S. Mays	EG&G, Idaho	V-II.A	October 1980
		V-II.B	October 1980
		VI-10.B	November 1979
		VII-3	June 1981
M. E. Nitzel	EG&G, Idaho	III-6	November 1979
E. Roberts	EG&G, Idaho	VIII-3.A	December 1979

*No longer with the Nuclear Regulatory Commission.

Name	Company	Topic	Report Date
A. Udy	EG&G, Idaho	VIII-1.A	March 1982
J. H. Cooper	EG&G, San Ramon	VI-4	April 1981
D. J. Morken	EG&G, San Ramon	VII-1.A	July 1981
		VIII-4	October 1981
B. M. Shindell	EG&G, San Ramon	VIII-2	October 1979
R. P. Kennedy	Engineering Decision Analysis Company	III-6	April 1980
R. Agarwal	Franklin Research Center	III-2	June 1982
		VI.10.A	January 1982
D. Barrett	Franklin Research Center	III-2	June 1982
L. Berkowitz	Franklin Research Center	III-1	March 1982
T. J. DelGazio	Franklin Research Center	VI-6	June 1981
A. Gonzales	Franklin Research Center	III-1	March 1982
R. Herrick	Franklin Research Center	IX-5	January 1982
J. E. Kaucher	Franklin Research Center	VI-6	June 1981
T. Stilwell	Franklin Research Center	III-7.B	December 1981
S. Tikoo	Franklin Research Center	III-1	March 1982
D. Bernreuter	Lawrence Livermore National Laboratory	II-4	October 1981
		II-4.A	October 1981
		II-4.C	October 1981
R. P. Rumble	Lawrence Livermore National Laboratory	VIII-2	October 1979
G. St. Leger-Barter	Lawrence Livermore National Laboratory	VI-7.A.3	November 1980
		VI-10.A	November 1980
		VII-2	November 1980
W. Stein	Lawrence Livermore National Laboratory	VI.2.D	November 1981
		VI-3	November 1981
F. J. Tokarz	Lawrence Livermore National Laboratory	III-6	April 1980
D. Vreeland	Lawrence Livermore National Laboratory	VI-2.D	November 1981
		VI-3	November 1981
W. J. Hall	Nathan M. Newmark Consulting Engineering Services	III-6	April 1980
N. M. Newmark	Nathan M. Newmark Consulting Engineering Services	III-6	April 1980
R. Spulak	Sandia National Laboratory	PRA	October 1982
P. Amico	Science Applications, Inc.	PRA	October 1982
D. Gallagher	Science Applications, Inc.	PRA	October 1982
J. McDonald	Texas Tech University	II-2.A	May 1980
J. Scherrer	Westec	II-3.A	May 1982
		II-3.B	May 1982
		II-3.B.1	May 1982
		II-3.C	May 1982
J. D. Stevenson	Woodward-Clyde Consultants	III-6	April 1980

APPENDIX H

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS COMMENTS ON DRAFT NUREG-0823 AND STAFF RESPONSE

UNITED STATES
NUCLEAR REGULATORY COMMISSION
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

December 13, 1982

Honorable Nunzio J. Palladino
Chairman
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Palladino:

SUBJECT: ACRS REPORT ON THE SYSTEMATIC EVALUATION PROGRAM REVIEW OF THE DRESDEN NUCLEAR POWER STATION, UNIT 2

During its 272nd meeting, December 9-11, 1982, the ACRS reviewed the results of the Systematic Evaluation Program (SEP), Phase II, as it has been applied to the Dresden Nuclear Power Station, Unit 2. These matters were also discussed during Subcommittee meetings in Washington, D. C. on October 27 and November 30, 1982. During our review, we had the benefit of discussion with representatives of the Commonwealth Edison Company (Licensee) and the NRC Staff. We also had the benefit of the documents listed below.

The Committee has reported to you previously on reviews of the SEP evaluations of the Palisades, Ginna, and Oyster Creek plants in letters dated May 11, August 18, and November 9, 1982. The first of these reports included comments on the objectives of the SEP and the extent to which they have been achieved. Our review of the SEP in relation to the Dresden plant has led to no changes in our previous findings regarding this program, as reported in our letter on the Palisades plant.

The remainder of this letter relates specifically to the SEP review of the Dresden plant.

Of the 137 topics to be addressed in Phase II of the SEP, 30 were not applicable to the Dresden plant and 19 were deleted because they were being reviewed generically under either the Unresolved Safety Issues (USI) program or the TMI Action Plan. Of the 88 topics addressed in the Dresden review, 54 were found to meet current NRC criteria or to be acceptable on another defined basis. We have reviewed the assessments and conclusions of the NRC Staff relating to these topics and have found them appropriate.

The 34 remaining topics involved 72 issues relating to areas in which the Dresden plant did not meet current criteria. These issues were addressed by the Integrated Plant Safety Assessment, and various resolutions have been proposed.

The Integrated Assessment has not yet been completed for 26 of the issues, for which the Licensee has agreed to provide the results of studies, analyses, and evaluations needed by the NRC Staff for its assessments and decisions. All of these issues are of such a nature that hardware backfits may be required for their resolution. The resolution of these issues will be addressed by the NRC in a supplemental report that will be available for review in connection with the application for a full term operating license (FTOL) for the Dresden plant.

For 21 of the issues included in the Integrated Assessment, the NRC Staff concluded that no backfit is required. We concur.

For the remaining issues for which the assessment has been completed, the NRC Staff requires hardware backfits in about half of the cases, and changes in procedures or Technical Specifications in the other half. The Licensee has agreed in all cases to make these changes.

As was the case for the Palisades, Ginna, and Oyster Creek plants, a plant-specific probabilistic risk assessment (PRA) was not available for the Dresden plant. In its place, the NRC Staff utilized the results of the Millstone Unit 1 PRA developed as part of the Integrated Reliability Evaluation Program (IREP), suitably modified and interpreted to reflect the differences between the two plants. The PRA study was used to address 19 of the issues included in the Integrated Assessment for the Dresden plant.

Our conclusions regarding the Dresden SEP review are similar to those for the plants previously reviewed:

1. The SEP has been carried out in such a manner that the stated objectives have been achieved for the most part for the Dresden plant and should be achieved for the remaining plants in Phase II of the Program.

2. The actions taken thus far by the NRC Staff in its SEP assessment of the Dresden plant are acceptable.

3. The ACRS will defer its review of the FTOL for the Dresden Nuclear Power Station, Unit 2 until the NRC Staff has completed its actions on the remaining SEP topics and the USI and TMI Action Plan items.

Sincerely,

P. Shewmon

P. Shewmon
Chairman

References:

1. U.S. Nuclear Regulatory Commission Draft Report, NUREG-0823, "Integrated Plant Safety Assessment, Systematic Evaluation Program, Dresden Nuclear Power Station, Unit 2," dated October 1982.
2. U.S. Nuclear Regulatory Commission Safety Evaluation Reports, Dresden 2 Systematic Evaluation Program Topics, Volumes 1 through 3, received October 1982.
3. NRC Staff consultants' reports on the Dresden 2 Integrated Plant Safety Assessment Report consisting of consultants' reports from S. H. Bush, J. M. Hendrie, H. S. Isbin and Z. Zudans, dated November 19, November 29, November 23, and November 24, 1982, respectively.
4. Science Applications, Inc. report number SAI-002-82-BE, "Interim Reliability Evaluation Program: Analysis of the Millstone Point Unit 1 Nuclear Power Plant," Volume I, Main Report, Draft dated October 1, 1982.

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

DEC 2 9 1982

Dr. Paul S. Shewmon, Chairman
Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Dr. Shewmon:

In your letter to Chairman Palladino dated December 13, 1982, the ACRS presented its views on the Systematic Evaluation Program Integrated Assessment Report for the Dresden Nuclear Power Station, Unit 2. In summary, this letter supported all of the staff's position.

Since the ACRS meeting on December 9, 1982, the staff has received a letter from Commonwealth Edison Company outlining their position on the open topics summarized in the Integrated Assessment. The licensee's letter confirms the staff presentation that there are no areas of disagreement between the staff and licensee.

The staff will revise a draft NUREG-0823 to reflect additional information provided by the licensee and respond to the recommendations and comments made by the staff's consultants.

Sincerely,

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

APPENDIX I

CONSULTANTS' COMMENTS ON DRAFT NUREG-0823 AND STAFF RESPONSE