NUREG–1403

Safety Evaluation Report

related to the full-term operating license for Dresden Nuclear Power Station, Unit 2

Docket No. 50–237

Commonwealth Edison Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

October 1990

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The Safety Evaluation Report for the full-term operating license application filed by Commonwealth Edison Company for the Dresden Nuclear Power Station, Unit 2, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in Grundy County, Illinois. Subject to favorable resolution of the items discussed in this report, the staff concludes that the facility can continue to be operated without endangering the health and safety of the public.

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1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

This report is a Safety Evaluation Report (SER) on the application for a full-term operating license (FTOL) for the Dresden Nuclear Power Station, Unit 2 (Dresden 2), based on an application filed by Commonwealth Edison Company (CECo), the licensee. This report was prepared by the U.S. Nuclear Regulatory Commission (the staff) and summarizes the results of the staff's review of the proposed conversion from a provisional operating license (POL) to an FTOL.

From 1959 to 1971, the Atomic Energy Commission issued POLs to 15 power reactors for periods up to 18 months as an intermediate stage before issuing an FTOL. The POL was issued to provide an interim period of routine operation during which the licensee and staff could assess plant operating parameters and performance against predicted values and resolve generic concerns identified during the licensing process. Thirty days after March 30, 1970, a rule change went into effect which deleted from the regulations the option of issuing POLs, but made no provision for converting previously issued POLs. Pursuant to 10 CFR 2.109, the POL would not be deemed to have expired provided the licensee filed an application for renewal at least 30 days before the expiration date. Since each of the POL licensees submitted a timely action for an FTOL, these POLs could continue indefinitely until the Commission completes its licensing action. Not withstanding the silence of regulations on conversion, the NRC policy has been to proceed with the POL conversion reviews. Only four of the POLs, including Dresden 2, remain to be converted.

In a letter dated November 15, 1972, CECo filed an application to convert POL DPR-19 for Dresden 2 to an FTOL. The facility received its POL on December 22, 1969. The POL review is documented in a safety evaluation forwarded to the licensee by letter dated October 21, 1969. The policy with respect to conversion of POLs to FTOLs was presented in SECY 83-19.

In 1975, because of a large backlog of unresolved generic issues that were relevant to the operation of the POL plants, the staff stopped its review of the POL conversions and set out to establish the appropriate scope of review needed to support the full-term conversion.

In 1977, the NRC staff recommended to the Commission that POL facilities be included in Phase II of the Systematic Evaluation Program (SEP) because much of the review necessary for converting the POLs was similar to the scope of the review proposed for the SEP. That recommendation was adopted.

The major portion of the technical support for the application for an FTOL for Dresden 2 comes from the SEP topic evaluations and SEP Integrated Plant Safety Assessment Report (IPSAR) and its supplement (NUREG-0823 and Supplement 1 to NUREG-0823). Since these issues have been extensively addressed in the IPSAR, only the topics that remain open are specifically discussed in this SER.

Facility improvements having safety significance have been submitted to the staff for formal review through letters and are documented in the updated Final Safety Analysis Report. Where required, the licensee has submitted proposed amendments to the Technical Specifications. Facility improvements approved by the staff have been inspected by the resident and/or regional inspectors to ensure they have been installed and operate as designed. The licensee has reviewed all plant modifications in accordance with 10 CFR 50.59. To ensure that the safety significance of changes or modifications made to the plant have been properly evaluated by the licensee, the staff periodically audits these reviews. On the basis of these efforts, the staff believes that major modifications made by the licensee have been adequately reviewed, approved, documented, and inspected by the staff and they need not be further addressed in this SER.

This SER addresses the remaining open issues generated by the accident at Three Mile Island Unit 2, as well as significant open multiplant actions for Dresden 2. Consideration has also been given to plant-specific open issues, such as proposed amendments to the Technical Specifications, and those determined to be significant have been addressed.

In accordance with the provisions of the National Environmental Policy Act of 1969, the staff prepared the draft and final environmental statements that set down the considerations related to the proposed POL-to-FTOL conversion. The Final Environmental Statement (FES) was issued in November 1973. Because the FES was issued a number of years ago, the staff performed an environmental assessment (EA) to determine if an FES supplement was necessary. The EA issued June 7, 1990, concluded that an FES supplement is not necessary.

The NRC Project Manager assigned to the FTOL review for Dresden 2 is Mr. Byron L. Siegel. Mr. Siegel may be contacted by calling (301) 492-3019 or by writing to U.S. Nuclear Regulatory Commission, ATTN: Byron L. Siegel, Office of Nuclear Reactor Regulation, Washington, D.C. 20555.

Availability of all material cited is given on the inside cover of this report.

1.2 Description

The Dresden Nuclear Power Station, Unit 2, located in Grundy County, Illinois, is a boiling-water reactor (BWR) designed by the General Electric Company. The licensee is the Commonwealth Edison Company. 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 FSAR was filed on November 17, 1967, and the initial POL was issued on December 22, 1969. In November 1972, the licensee applied for an FTOL. The licensed power rating currently is 2527 megawatts-thermal (MWt). The Dresden 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 BWR producing steam for direct use in the steam turbine. The reactor vessel contains internal components, including the equipment for separating steam and water flow paths.

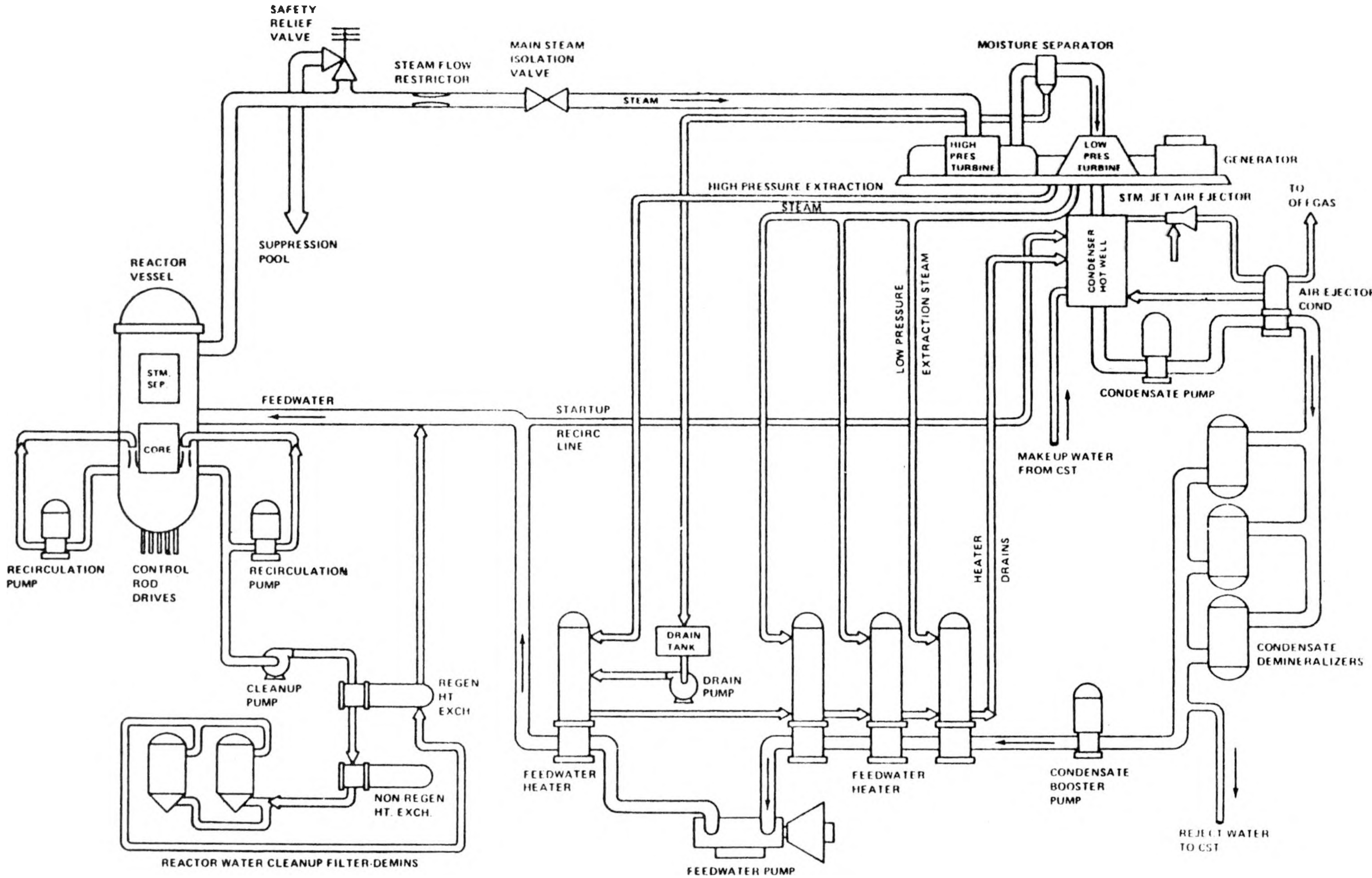

Figure 1.1 Dresden Unit 2 primary coolant system

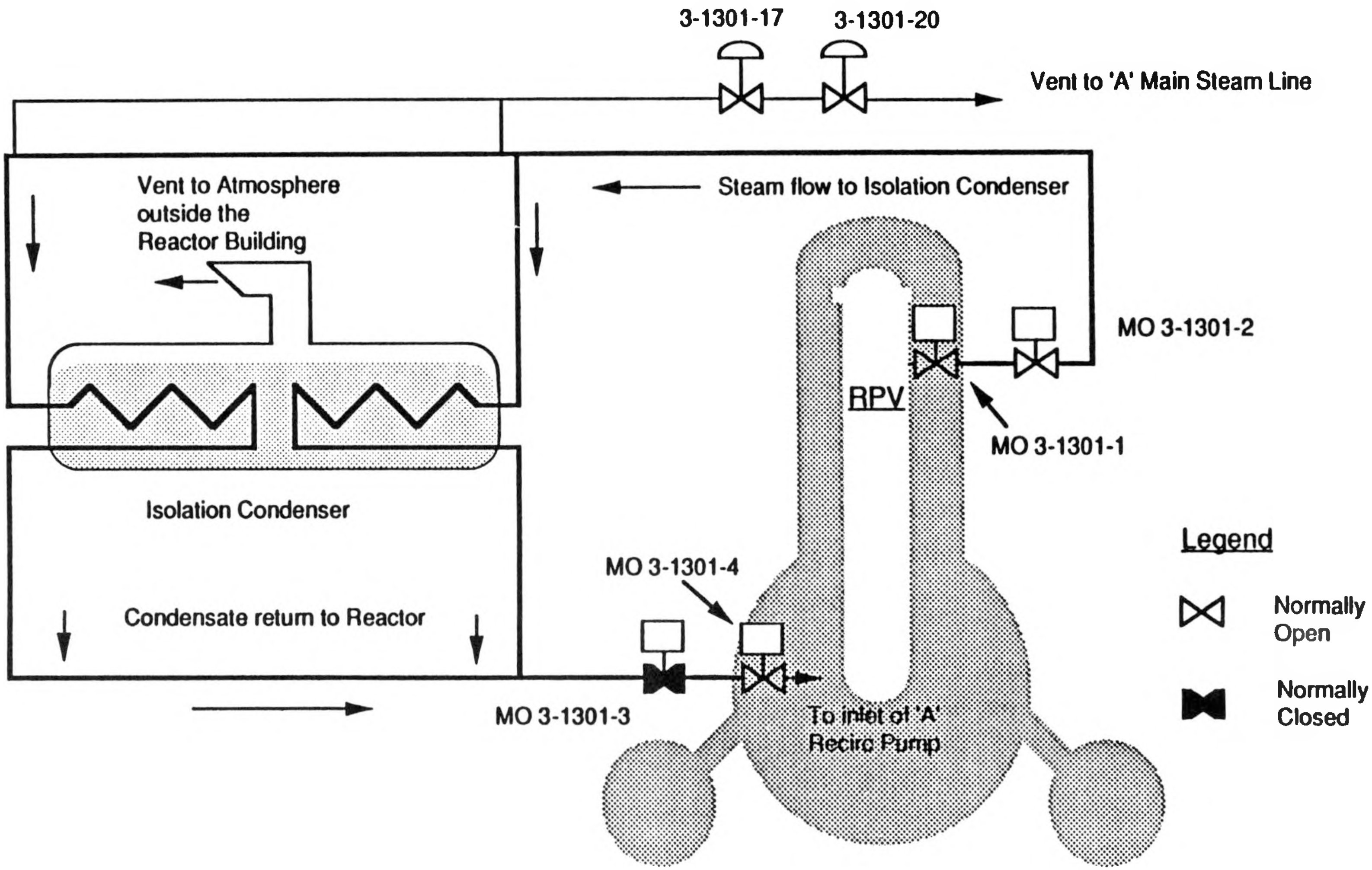

Figure 1.2 Isolation condenser subsystem

The recirculation system provides for forced flow through the reactor core to facilitate heat removal. The system consists of two external loops with motor-driven centrifugal pumps and 20 jet pumps located in the reactor vessel. Reactor coolant water mixed 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 mixture of steam and water.

The main steam system directs the steam generated in the reactor vessel to the turbine generator for conversion to electrical power. The steam and water mixture travels from the reactor core and through the steam-separating equipment. The steam passes into the four main steamlines and the water returns to the vessel. The steam then passes through the main steamlines to the turbine. Included in the main steam system are four relief, one safety/relief, and eight safety valves that 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 low-pressure emergency cooling systems will function. The relief valves, and the safety/relief valve are located upstream of the first isolation valve and discharge directly to the pressure-suppression pool; the safety valves are located on the steamlines 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 an incandescent light bulb and is constructed of steel plate, varies in diameter from 37 feet to 66 feet and is approximately 112 feet high. The shell thickness varies from approximately 3/4 to 2-3/4 inches. The pressure-suppression chamber is a steel pressure vessel in the shape of a torus with an inside diameter of 30 feet, a water volume of approximately 112,000 ft^3, and an air volume of approximately 117,000 ft^3.

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 (1) the monolithic reinforced-concrete floors and walls enclosing the nuclear reactor, primary containment, and reactor auxiliaries, and (2) the building superstructure with sealed panel walls and precast concrete roof.

2 SYSTEMATIC EVALUATION PROGRAM

The U.S. Nuclear Regulatory Commission (NRC) initiated the Systematic Evaluation Program (SEP) 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 compares 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 converting provisional operating licenses to full-term operating licenses.

Of the original 137 SEP topics, 49 were deleted from consideration within the SEP because a review was being made under other programs (unresolved safety issues or Three Mile Island Action Plan tasks), or the topic was not applicable to boiling-water reactors. The remaining 88 were reviewed for the Dresden Nuclear Power Station, Unit 2 (Dresden 2) and of these, 54 met current criteria or were acceptable on another defined basis. The licensee made no modifications during topic review. References for correspondence pertaining to safety evaluations for each of the 88 topics appear in Appendix E of the SEP Integrated Plant Safety Assessment Report (IPSAR, NUREG-0823).

The review of the remaining 34 topics revealed 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 called upon as well as the results of a limited probabilistic risk assessment study.

In general, the staff's positions resulting from the evaluations of the integrated assessment fell into one or more of the following categories: (1) equipment modification or addition, (2) procedure development or changes to Technical Specifications, (3) refined engineering analysis or continuation of evaluations already in process, and (4) no modification necessary. Table 4.1 of the IPSAR summarizes the staff's integrated assessment positions and documents the licensee's agreement with those positions as of February 1983.

For those positions classified above in either category 1 or 2, the IPSAR listed the scheduled completion dates agreed upon by the staff and the licensee.

For those positions in category 3, the licensee provided the results of the ongoing evaluation to the staff for review. For Dresden 2, 25 issues under 14 SEP topics either required refined engineering analysis or were continued under an ongoing evaluation. The evaluation of these issues and their status is summarized in Table 2.1 of Supplement 1 to the IPSAR. All but three of the issues identified in the IPSAR have been closed in the supplement. The staff is reviewing these remaining open issues.

2.1 SEP Topics

Topic III-1 Classification of Structures, Components, and Systems (Seismic and Quality) (IPSAR Section 4.2, Supplement 1 Section 2.1.2)

The SEP plants, including Dresden 2, were generally designed and constructed between the late 1950s and the late 1960s according to codes and criteria in effect at that time. Since then, however, the codes and criteria have been revised to incorporate the results of additional research. Thus, earlier plants may have been designed according to criteria and codes no longer accepted by the NRC. The purpose of SEP Topic III-1 is to compare the classification of structures, systems, and components of the as-built plants to the requirements in later editions of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code. The staff reviewed the earlier plants to determine whether their materials met the fracture toughness requirements in the ASME Code, Section III, 1977 Edition, Summer 1978 Addenda. The staff determined that fracture toughness requirements could differ significantly between the as-built Dresden 2 plant and the requirements in later editions of the ASME Code. The licensee was asked to determine which components could require impact testing to meet the fracture toughness requirements of the later editions of the ASME Code.

In a letter dated July 16, 1982, the licensee determined that impact testing could be required to determine the fracture toughness of the following components:

- core spray system pump casing
- low-pressure coolant injection (LPCI) pump casing
- LPCI heat exchangers--shell side
- high-pressure coolant injection (HPCI) pump casings

- HPCI piping, fittings, and valves with nominal pipe diameter greater than 6 inches

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

- main steam piping, valves, and fittings

In Enclosure 5 to a January 19, 1983 letter, the staff indicated to the licensee that compliance with the fracture toughness requirements of later editions of the ASME Code could be demonstrated by one of the following:

- providing test results that meet the ASME Code requirements

- determining that the component's lowest service temperature (LST) is high enough to exempt the materials from testing

- determining that the component's failure will not result in unacceptable consequences

In letters dated April 20, 1987, and January 6, 1989, the licensee provided information to demonstrate that the components identified in its July 16, 1982 letter, would meet the fracture toughness requirements of later editions of the ASME Code.

The staff has concluded that based on the information provided by the licensee, all components, except for the shell side of the LPCI heat exchangers, have adequate fracture toughness. The materials on the shell side of the LPCI heat exchangers will have adequate fracture toughness if their LST exceeds 77°F. In a letter dated May 1, 1989, the staff requested that the licensee determine whether the LST exceeds 77°F for the shell side of the heat exchanger. The licensee was also asked to identify (1) the operating conditions when the LST does not exceed 77°F, (2) the LST during these operating conditions, and (3) the design changes necessary so the LST exceeds 77°F. By letter dated March 30, 1990, the licensee provided the requested information; the staff is currently reviewing that information.

Topic III-6 Seismic Design Considerations (IPSAR Section 4.9)

In IPSAR Section 4.9.2(2), the staff stated it lacked sufficient information to evaluate the structural integrity of the reactor vessel and internal supports for Dresden 2. The staff also stated it will review the analyses of the Oyster Creek reactor vessel and internal supports to determine applicability to the Dresden 2 design. The staff has completed its review of the Oyster Creek analysis (March 12, 1990, letter to Oyster Creek) and is in the process of evaluating its applicability to Dresden 2.

III-7.B Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria (IPSAR Section 4.10, Supplement 1, Section 2.7)

10 CFR Part 50 (General Design Criteria 1, 2, and 4) as implemented by Standard Review Plan Section 3.8 (NUREG-0800), requires that structures be designed for the loadings that may be imposed on them and that they conform to applicable codes and standards.

Code, load, and load combination changes affecting specific structural elements have been identified wherein safety margins may be reduced from those required by current standards. Therefore, the staff position in the IPSAR was that the licensee should provide information regarding the applicability of the code changes and the safety margins.

The licensee submitted this information in letters dated August 2, 1982, and July 11, 1984. The staff's contractor, Franklin Research Center, reviewed the design code changes and load combination issues and issued a technical evaluation report (TER) (C5506-425) dated June 3, 1986. Section 6 of the TER identifies items not addressed, items which were addressed by the licensee using a sampling philosophy, and items requiring further clarification. By letter dated July 26, 1989, the staff requested additional information to close these open items identified in the TER. The licensee, by letter dated August 30, 1989, provided the requested information which the staff is reviewing.

2.2 IPSAR Topic Resolution Confirmed by the NRC Region III Office

During the integrated assessment program for Dresden 2, a number of issues were resolved by commitments made by the licensee for specific plant modifications or procedural changes. The NRC Region III Office was asked to confirm that these commitments have been implemented.

The Region III staff conducted onsite inspections for each item identified in Table 2.1 as complete or partially complete. (This is an update of Table 4.1 in Supplement 1 to the IPSAR.) The Region III staff examined installed equipment and reviewed supporting procedures and other documentation. The Region III staff concluded that the licensee had met the commitments documented in the IPSAR for those items in Table 2.1 whose status is designated as "complete." Inspection findings with the results of these reviews are documented in the inspection reports identified in Table 2.1. The remaining items identified as open will be closed in future inspection reports.

Table 2.1 Items for confirmation by NRC Region III Office

Item No.	Description of confirmation	SEP/IPSAR reference	Status of dockets 50-237 & 50-249 (Inspection Reports)
(1)	Install scuppers to prevent ponded water on roofs.	Topic II-3.B/ 4.1.3	Complete (85-30)
(2)	Revise emergency plan to cope with design-basis flooding.	Topic II-3.B.1/ 4.1.4	Partial (85-30) Complete (89018 and 89019)
(3)	Modify procedures for inspecting water control structures.	Topic III-3.C/ 4.4.3	Complete (83-32)

Table 2.1 (Continued)

Item No.	Description of confirmation	SEP/IPSAR reference	Status of dockets 50-237 & 50-249 (Inspection Reports)
(4)	Review containment penetrations and provide locking devices and administrative control procedures as necessary.	Topic VI-4/ 4.18.1	Partial (83-32) Complete (89018 and 89019)
(5)	Lock valves closed and modify procedures for manual isolation valves identified.	Topic VI-4/ 4.18.3	Partial (85-30) Complete (89018 and 89019)
(6)	Provide second locked closed isolation valve on identified lines.	Topic VI-4/ 4.18.6	Partial (85-30) Redundant valve installation not confirmed
(7)	Provide procedures to ensure disconnect links are properly positioned following maintenance.	Topic VI-7.C.1/ 4.21.2	Complete (89018 and 89019)
(8)	Provide procedures to address use of breakers 252-2829 during power operation.	Topic VI-7.C.1/ 4.21.3	Complete (83-32)
(9)	Install Class 1E protection between reactor protection system (RPS) power supply and RPS.	Topic VII-1.A/ 4.24.3	Complete (83-32)
(10)	Implement procedures for testing shutdown cooling system temperature interlocks.	Topic VII-3/ 4.25.4	Complete (83-32)
(11)	Bypass diesel generator underfrequency protective trips during emergency operations.	Topic VIII-2/ 4.26.2	Partial (85-30) Complete (89018 and 89019)
(12)	Provide monitoring of dc system in control room.	Topic VIII-3.B/ 4.28	Partial (85-30) Complete (89018 and 89019)
(13)	Confirm that the plant procedures adequately address alternate means of shutdown if components not enclosed in qualified structures are lost as a result of wind and tornado loadings.	Topic III-2/ 2.2.2 (Supp. 1)	Open

Table 2.1 (Continued)

Item No.	Description of confirmation	SEP/IPSAR reference	Status of dockets 50-237 & 50-249 (Inspection Reports)
(14)	Confirm that the licensee has the equipment necessary on site to repair or remove the damaged components of the diesel generators.	Topic III-4.A/ 4.5.3 and 2.2.2 (Supp. 1)	Open
(15)	Confirm that the licensee has implemented a plant-specific analysis of the structural integrity of cable trays to ensure their ability to maintain their integrity and that the cable tray support systems have been modified where necessary.	Topic III-6/ 4.9.3	Open
(16)	Confirm that the licensee has installed proper leak-rate test taps on the reactor building closed cooling water lines.	Topic VI-4/ 4.18.2; Topic VI-6/ 4.19	Open
(17)	Confirm that the leakage conditions under which the remote manual isolation valves on the LPCI and core spray systems should be isolated are incorporated into the emergency procedures.	Topic VI-4/ 4.18.2 and 2.10 (Supp. 1)	Open
(18)	Confirm that the licensee no longer permits paralleling of the 125-V dc and 250-V dc systems during power operation of either Dresden unit and that ground detection procedures for both units have been revised accordingly.	Topic IV-10.B/ 2.12 (Supp. 1)	Open
(19)	Confirm that operating procedures have been changed to require a "normal-normal" alignment of the generator 2/3 normal/bypass switches.	Topic VI-10.B/ 4.23.2	Open

Table 2.1 (Continued)

Item No.	Description of confirmation	SEP/IPSAR reference	Status of dockets 50-237 & 50-249 (Inspection Reports)
(20)	Confirm that the licensee has installed Class 1E signal isolation devices at the inputs of each control room recorder that monitor the RPS as committed to in the January 9, 1987, and February 2, 1989, letters.	Topic VII-1.A/ 4.24.1 and 2.13.1 (Supp. 1)	Open
(21)	Confirm that the licensee has provided isolation between the average power range monitors and the process computer by the installation of "flying" capacitors."	Topic VII-1.A/ 4.24.2 and 2.13.2 (Supp. 1)	Open
(22)	Confirm that the bypass diesel generator underfrequency trip modifications are completed for diesel generator 2/3.	Topic VIII-2/ 4.26.2	Open

3 THREE MILE ISLAND "LESSONS LEARNED" REQUIREMENTS

As a result of the accident at Three Mile Island Unit 2 (TMI-2), the staff of the U.S. Nuclear Regulatory Commission (NRC) considered recommendations made by groups that were established to investigate the accident, and developed new requirements to improve the regulation and operation of nuclear facilities. These groups included:

- Congress
- General Accounting Office
- President's Commission on the Accident at Three Mile Island
- NRC Special Inquiry Group
- NRC Advisory Committee on Reactor Safeguards
- Lessons Learned Task Force
- Special Review Group of the NRC Office of Inspection and Enforcement
- NRC Siting Task Force
- NRC Emergency Preparedness Task Force
- NRC Office of Standards Development
- NRC Office of Nuclear Regulatory Research.
- Bulletins and Orders Task Force of the NRC Office of Nuclear Reactor Regulation

NUREG-0660, entitled "NRC Action Plan Developed as a Result of the TMI-2 Accident" (referred to as the Action Plan), was developed to provide a comprehensive and integrated plan for the actions NRC judged necessary to correct or improve the regulation and operation of nuclear facilities. The Action Plan was based on the experience from the TMI-2 accident and the recommendations of the investigating groups.

With the development of the Action Plan, NRC transformed the recommendations of the investigating groups into discrete scheduled tasks that specify changes in regulatory requirements, organization, or procedures. Some actions to improve the safety of operating plants were judged to be necessary before an action plan could be developed, although they were subsequently included in NUREG-0660. Such actions came from the bulletins and orders issued by the Commission immediately after the accident, the first report of the Lessons Learned Task Force, and the recommendations of the Emergency Preparedness Task Force. Before these immediate actions were applied to operating plants, they were approved by the Commission.

NUREG-0737, entitled "Clarification of the TMI Action Plan Requirements," was issued in November 1980. This report identified the specific items from NUREG-0660 that were approved by the Commission for implementation at nuclear power plants. It also included additional information about schedules, applicability, method of implementation review, submittal dates, and clarification of technical positions. By letter dated December 17, 1982, Supplement 1 to NUREG-0737 was issued to coordinate and indicate initiatives related to:

- safety parameter display systems
- detailed control room design reviews
- application of Regulatory Guide 1.97 to emergency response facilities

- upgrading emergency operating procedures
- emergency response facilities
- emergency operations facility
- technical support center
- operational support center
- meteorological data

Of the TMI Action Plan requirements for boiling-water reactors documented in NUREG-0737, three have yet to be totally resolved for Dresden 2. The three open items are:

TMI Item	Title
I.D.1.1	Detailed Control Room Design Review
II.F.2.4	Instrumentation for Detection of Inadequate Core-Cooling Installation of Additional Instrumentation
III.A.2.2	Upgrade Emergency Preparedness - Meteorological Data
RG 1.97	Application to Emergency Response Facilities

I.D.1.1 Detailed Control Room Design Review

In a letter dated July 12, 1989, the staff sent its safety evaluation of the detailed control room design review (DCRDR) to the licensee. In its safety evaluation, the staff concluded that pending completion of outstanding commitments and corrective actions in regard to human engineering discrepancies (HEDs), the licensee satisfies the requirements of Supplement 1 to NUREG-0737.

In letters dated May 26, 1989 and November 17, 1989, the licensee provided the status of the DCRDR implementation for Dresden Station. The HEDs that remain to be completed are primarily related to recorder setpoints, numeral height, meter scale graduations and annunciator modifications. In the November 17, 1989 letter, the licensee requested schedular relief for the annunciator modifications for an additional operating cycle. The schedular relief was requested because the total man-hours for completion is much larger than previously anticipated and because of the limited physical space within each panel which restricts the size of the crew at any given time. The staff reviewed the licensee's submittal and determined that the licensee's request is acceptable based on the fact that the HEDs are Category 2, Level B or C, which are not safety significant and have only minimal influence on determining plant status during operating events, and that data to justify the time required to make these modifications are available from Quad Cities.

II.F.2.4 Instrumentation for Detection of Inadequate Core Cooling--Installation of Additional Instrumentation

The remaining requirement of Generic Letter 84-23 related to reactor vessel instrumentation in boiling-water reactors for Dresden 2 to satisfy all the requirements of TMI-2 Item II.F.2 is: "Improvements to plant(s) that will reduce level indications errors caused by high drywell temperature. These improvements include prevention of reference leg overheating or reduction of the vertical drops in the drywell."

By letter dated August 3, 1989, the licensee stated that the implementation of the modification for rerouting the reactor water reference leg for Dresden 2 is scheduled to be completed during the next refueling outage (September 1990).

III.A.2.2 Upgrade Emergency Preparedness--Meteorological Data

TMI-2 Item III.A.2 required each nuclear facility to upgrade its emergency plans to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Specific criteria to meet this requirement are delineated in NUREG-0654, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparation in Support of Nuclear Power Plants."

Revision 1 to NUREG-0654 provides meteorological criteria to fulfill, in part, the standard that, "Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use" (see 10 CFR 50.47). The position in Appendix 2 to NUREG-0654 outlines four essential elements that can be categorized into three functions: measurements, assessment, and communications.

In a letter dated January 31, 1989, the licensee provided a revised schedule of May 30, 1990 for implementing the A model for calculating meteorological data at Dresden 2. The schedule was revised because: quality assurance enhancements require more formalized validation and verification of computer software; numerous scope changes have resulted in additional changes; and resource limitations. In a letter dated March 30, 1990, the licensee further revised its target date for completing the A model implementation at Dresden 2 to May 31, 1991. In the interim, the licensee has the capability to evaluate the meteorological data using alternate methods.

RG 1.97 Application to Emergency Response Facilities

Regulatory Guide (RG) 1.97 and 10 CFR 50.49 require that neutron flux be monitored by Category 1 instrumentation. The staff has allowed boiling-water reactors to operate with existing instrumentation until instrumentation was developed that conforms to the requirements of RG 1.97 and 10 CFR 50.49.

By letter dated April 1, 1988, the Boiling Water Reactor Owners Group (BWROG) submitted General Electric Report NEDO-31558 which proposed functional criteria for post-accident neutron flux monitoring as an alternative to the Category 1 instrumentation recommendations specified in RG 1.97. The BWROG-proposed alternative position, which was determined to be unacceptable, is contained in the staff's safety evaluation report that was transmitted to the licensee by letter dated February 14, 1990.

Based on the understanding that instrumentation that meets the requirements of RG 1.97 and 10 CFR 50.49 is available, the staff, in a letter dated February 14, 1990, requested that CECo provide a schedule for installation of neutron flux monitoring instrumentation that meets the requirements of these documents for the Dresden plants. Since that time the BWROG has interceded (February 21, 1990 letter) and raised several generic issues associated with the design requirements of this instrumentation. The staff and the BWROG are working towards obtaining resolution of these issues (May 21, 1990 letter). When this has been achieved, CECo will be requested to implement this generic resolution on a plant specific basis.

4 SIGNIFICANT OPEN ISSUES

The staff has evaluated the open issues on the Dresden 2 docket. These issues can be primarily categorized as follows:

- licensing amendments
- plant-specific resolution of multiplant actions
- generic letters
- bulletin responses
- plant-specific issues requiring resolution
- unresolved safety issues which have a generic resolution but have not been resolved on a plant-specific bases.

On the basis of this evaluation, the staff has identified the open issues on the Dresden 2 docket which have safety significance. These issues are discussed below.

4.1 Intergranular Stress Corrosion Cracking (Generic Letter 88-01)

Intergranular stress corrosion cracking (IGSCC) at welds in boiling-water reactor (BWR) piping has been of continuous concern for almost 20 years. An ever-increasing amount of research and developmental activity related to understanding the causes of the cracking and ways to prevent it has been in progress during this period. Under the auspices of the NRC, two Pipe Crack Study Groups have reviewed the problem in BWRs--one in 1975 and the other in 1979. The findings of these groups were published in NUREG-75/067, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," and NUREG-0531, "Investigation and Evaluation of Stress Corrosion Cracking in Piping of Light Water Reactor Plants." The staff guidelines to implement the recommendations contained in these reports were published in NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," and NUREG-0313, Revision 1.

NUREG-0313 was first revised in 1980 to provide guidance and recommendations regarding materials and processes that could be used to minimize IGSCC and to provide recommendations regarding the augmentation of the extent and frequency of inservice inspections of welds considered to be susceptible to IGSCC. Revision 1 also provided recommendations regarding the upgrading of leak detection systems and leakage limits for plants with susceptible welds.

In NUREG-0313, Revision 2, issued as an enclosure to Generic Letter 88-01, these recommendations were revised and several subjects were added. Revision 2 provides the following:

- guidance for performing American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, IWB 3600, calculations for flaw evaluation

- recommendations regarding the repair of cracked piping recommendations regarding formal performance demonstration tests for ultrasonic test examiners, such as those prescribed by IE Bulletins 82-03, "Stress Corrosion Cracking in Thick-Wall, Large-Diameter, Stainless Steel Recirculation System Piping at BWR Plants," and 83-02, "Stress Corrosion Cracking in Large-Diameter Stainless Steel Recirculation System Piping at BWR Plants." (These tests, which are currently being conducted under the Nondestructive Examination Coordination Plan, agreed upon by the NRC, the Electric Power Research Institute, and the BWR Owners Group, provide additional assurance that inspections for IGSCC in BWR piping are performed effectively.)

The approach used in previous revisions of NUREG-0313 to identify welds that require augmented inspection is simplified, but is expanded to include consideration of reinspections of welds found to be cracked, with or without repair or mitigative actions. The current approach is based on the following:

- All stainless steel welds in high-temperature BWR systems are considered to be subject to IGSCC unless measures have been taken to make them resistant.

- The frequency and sample size used to inspect all safety-related piping welds in BWR plants will depend on the material and processing used. Sample bases are provided for such classification.

- Some utilities may choose not to replace piping, or to operate for some interim period before making major modifications or replacing piping. Guidance is provided to cover these situations in which a utility chooses to operate with cracked or repaired welds.

The staff has reviewed the licensee's submittals dated February 6 and March 30, 1989, which included inspection results, IGSCC mitigation flaw evaluations, and overlay repairs to support the continued operation of Dresden 2.

Generic Letter 88-01 applies to a total population of 276 welds at Dresden 2. Of those, 228 are considered to be susceptible to IGSCC (not category A). A total of 192 welds were ultrasonically tested (UT) in the 1988 outage (eleventh refueling outage) including 190 (83 percent) of the 228 susceptible welds. A total of 104 (46 percent) of the 228 susceptible welds were mechanically stress improved (MSIP) this outage. All MSIP welds were inspected after the stress improvement.

Nineteen new flawed welds were identified this outage. Three welds with unrepaired circumferential indication reported during the previous outage were reexamined this outage. All of the circumferential flaws were reported as unchanged. However, two of the three welds were reported to contain new axial indications.

New weld overlay repairs were applied to 21 welds this outage. Also, three "leak barrier" weld overlays applied during a previous outage were built up to standard thickness, surface finished, and baseline (UT) examined this outage. Because so many weld overlay activities took place this outage, two-layer leak barrier weld overlays were applied as a temporary fix over two welds with axial indications only. These will be built up to standard thickness, surface finished, and examined during the next outage.

Dresden 2 has completed three consecutive cycles of hydrogen water chemistry (HWC) and the staff approved the licensee's request for a factor of two reduction in the inspection of category C, D, and E weldments in accordance with Generic Letter 88-01 for the 1988 refueling outage.

However, in view of the extensive IGSCC found in this outage, the staff has generic concerns regarding the effectiveness of HWC in mitigating the IGSCC. The staff noted that the HWC implemented in this unit is neither monitored by electrochemical potential measurements nor confirmed by on-line crack-arrest verification testing. One possible explanation for the reported inspection results is that the hydrogen injection rate might not be large enough to effectively mitigate the IGSCC. To ensure adequate inspection of IGSCC-susceptible piping welds, the staff has determined that any future request for reducing the scope or frequency of IGSCC inspections will not be granted until the staff has resolved its concern about the effectiveness of HWC in mitigating the IGSCC. The staff will evaluate the effectiveness of HWC at Dresden 2.

On the basis of its review of the licensee's submittals, the staff concludes that the licensee has adequately addressed IGSCC in Class 1 piping with respect to inspections, repairs, and mitigations performed during the 1988 refueling outage at Dresden 2, and that these activities were performed in accordance with the guidelines in Generic Letter 88-01. In addition, the staff also concludes that Dresden 2 can be safely operated for another 18-month fuel cycle in the present configuration and does not pose a threat to the health and safety of the public.

4.2 Control Room Habitability

NUREG-0737, "Clarification of TMI Action Plan Requirements," Task Action Plan Item III.D.3.4 ("Control Room Habitability"), requires that the operators in the control room be adequately protected against the effects of accidental releases of toxic and radioactive gases. This would ensure safe operation or shutdown under design-basis-accident conditions at Dresden 2.

The staff requested technical specifications (TS) related to control room habitability in Generic Letter 83-36, "NUREG-0737 Technical Specifications," dated November 1, 1983. In a letter dated February 9, 1984, in response to Generic Letter 83-36, the licensee stated the new control room heating, ventilation, and air conditioning (HVAC) system installation and testing have not been completed and that TS changes will be submitted after completion of the modification. In a letter dated August 1, 1988, the licensee submitted proposed TS changes to assure the availability and effectiveness of the new control room air filtration system that has been installed. This amendment is currently under staff review.

4.3 Combustible Gas Control (SEP Topic VI-5)

SEP Topic VI-5, "Combustible Gas Control," concerns the potential for combustible gas conditions (i.e., hydrogen production due to zirconium-water reaction, water radiolysis, and corrosion of metals after an accident). As amended on December 2, 1981, 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors," delineates the requirements pertaining to preventing the accumulation of combustible gases in the containment following design-basis accidents. Generic Letter (GL) 84-09, "Recombiner Capacity Requirements of

10 CFR 50.44(c)(3)(ii)," dated May 8, 1984, transmitted the Commission determination that inerted Mark I BWR containments (for which notices on the construction permits were published before November 5, 1970) need not rely on safety-grade purge/repressurization systems as the primary means of hydrogen control, provided the following criteria are met:

- The plant has TS (limiting conditions for operation) requiring that, when the containment is required to be inerted, the containment atmosphere be less than 4 percent oxygen.

- The plant has only nitrogen or recycled containment atmosphere for use in all pneumatic control systems within the containment.

- There is no potential source of oxygen in the containment other than that resulting from radiolysis of the reactor coolant.

In a letter dated June 25, 1984, the licensee responded to GL 84-09 and provided information supporting the conclusion that both containment recombiner capacity and a containment air dilution system currently installed, are not required. The staff's position with regard to combustible gas control is contained in a memorandum dated May 19, 1986. In this memorandum, the staff determined that the use of an air containment dilution system was unacceptable and identified the acceptable approaches for complying with 10 CFR 50.44 and GL 84-09. On January 20, 1987, the staff and utility representatives, including Commonwealth Edison Company (CECo), met to discuss the system used in their plants for combustible gas control. The staff sent a copy of the meeting summary, which describes the significant items discussed, to CECo in a letter dated April 24, 1987.

As a result of this meeting, the staff requested that each licensee submit its plant-specific position on its compliance to 10 CFR 50.44(g). This submittal should include the assumptions made by the licensees to justify their position on 10 CFR 50.44 and the information discussed during the meeting on the reliability and capability of the containment inerting system and the window of accident sequences for which this system would be effective in controlling combustible gases. The staff stated that a passive system, such as the inerted containment, is not sufficient to meet 10 CFR 50.44(g) and that an active system, such as the containment inerting system, is required. The staff further stated that the reliability and capability of the existing containment inerting systems may be sufficient to meet, as a minimum, the intent of General Design Criteria (GDC) 41, 42, and 43 and of 10 CFR 50.44(g). This is because the Regulatory Guide (RG) 1.7 hydrogen and oxygen source term indicative of large metal-water reactions may show that the licensee has sufficient time to respond with the existing system to the increasing combustible gas concentrations in the containment from radiolysis of water before the acceptable limits are exceeded.

Since the time available until unacceptable concentrations are reached would allow the licensee to overcome the lack of redundancy in components and in providing power to the system, the staff stated that the licensee should discuss this time period for the plant and the actions taken in the licensee's justification of the reliability of its containment inerting system.

The licensee has not yet provided the information requested by the staff. In an attempt to resolve this issue, the staff, in a letter to the licensee dated

May 3, 1989, requested that a meeting be held with CECo to review the current status of combustible gas control at Dresden and Quad Cities. This meeting has been postponed pending a legal determination by the staff that Oyster Creek, as it is currently designed, is in compliance with the requirements of 10 CFR 50.44.

4.4 Station Blackout

Electrical power for safety systems at nuclear power plants must be supplied by at least two redundant and independent divisions. The systems used to remove decay heat to cool the reactor core following a reactor shutdown are included among the safety systems that must meet these requirements. Each electrical division for safety systems includes two offsite alternating current (ac) power connections, a standby emergency diesel generator (EDG) ac power supply, and direct current (dc) sources.

Generic Task A-44 involves a study of whether or not nuclear power plants should be designed to accommodate a complete loss of all ac power; that is, a loss of both the offsite and the EDG power supplies. This issue arose because of operating experience regarding the reliability of ac power supplies. There have been numerous reports of EDGs failing to start and run in operating plants during periodic surveillance tests. In addition, a number of operating plants have experienced a total loss of offsite electrical power, and more are expected to report this in the future. In almost every one of these loss-of-offsite-power events, the onsite emergency ac power supplies were available immediately to supply the power needed by vital safety equipment. However, in some instances, one of the redundant emergency power supplies has been unavailable. In a few cases, there has been a complete loss of ac power, but during these events, ac power was restored in a short time without serious consequences.

A loss of offsite power involves a loss of both the preferred and backup sources of offsite power. If all offsite power is lost, the onsite emergency ac power system will provide ac power to safety-related equipment. With respect to emergency onsite ac power, the Dresden 2 emergency generators are powered by one dedicated diesel engine and one shared swing diesel engine. These systems have been evaluated under SEP Topic VIII-2 and found acceptable. The staff's evaluation is presented in the IPSAR (NUREG-0823), Section 4.26.

A loss of all ac power was not a design-basis event for Dresden 2. Nonetheless, a combination of design, operating, and testing requirements has been imposed to ensure that this facility will have substantial resistance to a loss of all ac power and that, even if all ac power should be lost, there is reasonable assurance the core will be cooled.

The current licensing criteria require licensees to provide redundant emergency ac power supplies, to demonstrate emergency ac power supply reliability (RG 1.108), and to include the capability of removing decay heat using at least one shutdown cooling train independent of ac power. Boiling-water reactors contain various systems to remove core decay heat following the total loss of ac power. These systems at Dresden 2 consist of a passive isolation condenser and a steam-driven HPCI which will allow time for restoration of ac power from either offsite or onsite sources.

On the basis of such considerations, the staff concludes that there is reasonable assurance that Dresden 2 can be operated before full compliance with the resolution of this generic issue is achieved without endangering the health and safety of the public.

On June 21, 1988, the Commission issued the Station Blackout Rule, 10 CFR 50.63, which resolves and supersedes Generic Task A-44. The Station Blackout Rule is implemented by Multiplant Action (MPA) Item A-22. Compliance with this MPA item is being achieved through normal licensing action.

The licensee submitted a response to the Station Blackout Rule by letter dated April 17, 1989. Working meetings were held with the licensee on October 4 and 5, 1989, December 20, 1989 and March 28, 1990 to discuss the licensee's responses for both Dresden and Quad Cities. Meeting summary memoranda dated November 9, 1989, March 5, 1990 and May 2, 1990 contain the staff's findings. In a letter dated May 18, 1990, the licensee provided a revised response to its station blackout submittal that resulted from the discussions and clarifications that evolved during these working meetings. In this revised submittal, the licensee proposed the following modifications:

- Installation of an alternate AC power source.
- Installation of a crosstie between Dresden Units 2 and 3 safety busses to improve the offsite power system.
- Logic changes allowing the shared emergency diesel generator to connect to Dresden Units 2 and 3 safety busses simultaneously from the control room.
- Installation of an isolation condenser level indication transmitter qualified for the expected station blackout thermal profile.

In addition, since the licensee's revised submittal proposes an alternate AC power source, the ability to withstand the loss of AC power has been reduced to 1 hour compared to the 4 hours required by the approach utilized in the initial submittal. Attachment A to the licensee's May 18, 1990 submittal addresses the ability of the Dresden Station to cope with a loss of AC power for this 1 hour period. The licensee's submittal and proposed implementation date of December 1995 are under staff review.

4.5 Hardened Wetwell Vent

As a part of a comprehensive plan for closing severe accident issues, the staff undertook a Mark I containment improvement program to determine if any actions should be taken, on a generic basis, to reduce the vulnerability of BWR Mark I containments to severe accident challenges. At the conclusion of this program, the staff identified a number of plant modifications that substantially enhance the plants' capability to both prevent and mitigate the consequences of severe accidents. The improvements that were recommended include (1) improved hardened wetwell vent capability, (2) improved reactor pressure vessel depressurization system reliability, (3) an alternative water supply to the reactor vessel and drywell sprays, and (4) updated emergency procedures and training.

The staff has been directed by the Commission to pursue these Mark I enhancements on a plant-specific basis in order to account for possible unique design differences that may bear on the necessity and nature of specific safety improvements. The Commission also determined that the recommended safety improvements, with one exception, that is, hardened wetwell vent capability, should be evaluated by licensees as part of the Individual Plant Examination (IPE) Program. With regard to the recommended plant improvement dealing with hardened vent capability, the Commission, in recognition of the circumstances and benefits associated with this modification, directed a different approach. Specifically, the Commission directed the staff to approve installation of a hardened vent under the provisions of 10 CFR 50.59 for licensees, who on their own initiative, elect to incorporate this plant improvement.

In response to the Commission's guidance, the staff issued Generic Letter 89-16 related to the installation of hardened wetwell vents. In this Generic Letter, the staff strongly encouraged licensees to install a hardened wetwell vent, utilizing portions of existing systems, to the greatest extent practical under the provisions of 10 CFR 50.59. The Generic Letter also stated that for facilities not electing to voluntarily install a hardened wetwell vent the staff will perform a plant specific backfit analyses. To more accurately reflect plant specificity each licensee not installing a vent was requested to provide cost estimates for implementation of a hardened vent by pipe replacement.

The licensee responded to Generic Letter 89-16 in a letter dated October 30, 1989 that was supplemented by a letter dated May 24, 1990. The licensee, in its initial response, stated Dresden 2 and 3 have isolation condensers that are capable of mitigating the loss of decay heat removal (TW) sequence that would require containment venting. The licensee also committed to provide the rationale for the use of isolation condensers in lieu of hardened vents. A cost estimate for installation of a hardened wetwell vent was also included. In its May 24, 1990 letter the licensee provided this rationale in the form of a comparison of the Dresden isolation condenser performance to the BWR Owner Group (BWROG) design criteria for a TW sequence of events.

The staff in a letter dated June 15, 1990 provided the licensee the results of the staff initiated backfit analyses using the licensee's plant-specific cost estimates. The staff estimated the benefits of venting by determining the reductions in core damage frequencies (CDFs) for the TW sequences. The benefits were calculated by using the results of the probabilistic risk assessments (PRAs) for BWRs with Mark I containments and isolation condenser systems (ICS) similar to Dresden Units 2 and 3. The staff then adjusted the analyses to account for recent advances in the PRA methodology (NUREG-1150). The results of the staff's analyses showed that for TW sequences alone the overall CDF for each of the Dresden Units 2 and 3 can be reduced by 1.4 E-5 per reactor year. The credit for the operation of the ICS was included in the analyses. The analyses were adjusted to account for the power levels of Dresden Units 2 and 3, and the density of population surrounding the Dresden site. The staff has calculated that for TW sequences alone, the operation of vent would avert the expected radiological exposure to public by 50.2 man-rem per reactor year. Using 20 years of remaining plant life for Dresden Unit 2, and 21 years of remaining plant life for Dresden Unit 3 and plant-specific modification costs, the staff has estimated an averted radiological population exposure for 1005 man-rem per million dollars for Unit 2 and 1055 man-rem per million dollars for Unit 3. The results of the staff analyses demonstrate that hardened vent capabilities would provide significant benefits in the expected reduction in radiological exposure risks posed by TW sequences.

On the basis of this analysis the staff requested that the licensee reconsider its decision and commit to install a hardened wetwell vent at Dresden Units 2 and 3. The June 15, 1990 letter also stated that in the absence of such a commitment the staff intends to pursue the imposition of a hardened wetwell vent under the provisions of the Commissions backfit rule (10 CFR 50.109). The staff is currently waiting for the licensee's response which is due in August 1990.

5 UNRESOLVED SAFETY ISSUES

In Generic Letter (GL) 89-21, dated October 19, 1989, the staff requested that each licensee provide the status of implementation of unresolved safety issues (USIs). In a letter dated November 29, 1989, the licensee responded to GL 89-21. In a memorandum for file dated February 21, 1990, the staff documented the status of the USIs applicable to Dresden 2 and 3. This memorandum included: a copy of the information provided by the licensee in its response to GL 89-21, a status summary for each USI applicable to Dresden, and a copy of the staff's data base printout for Dresden which reflects the staff's assessment of all the USIs applicable to Dresden.

Of the 27 USIs, 14 were determined to be applicable to Dresden 2 and 3. These were:

A-01 Water Hammer
A-06 Mark 1 Short-Term Program
A-07 Mark 1 Long-Term Program
A-09 Anticipated Transients Without Scram
A-10 Feedwater Nozzle Cracking
A-11 Reactor Vessel Materials Toughness
A-17 Systems Interaction
A-24 Qualification of Class 1E Safety-Related Equipment
A-36 Control of Heavy Loads
A-42 Pipe Cracks in Boiling-Water Reactors
A-44 Station Blackout
A-46 Seismic Qualification of Equipment in Operating Plants
A-47 Safety Implications of Control Systems
A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

Of the USIs applicable to Dresden, all but five are complete. Station Blackout (A-44) and Hydrogen Control (A-48) have been generically resolved and the status of the plant specific implementation is described in Section 4 of this report. The remaining three USIs are addressed below. Additional information on the incomplete USIs and the bases for resolution of those USIs which are complete can be obtained by consulting the February 21, 1990, memorandum for file.

USI A-09 Anticipated Transients Without Scram

The licensee has installed all modifications to meet the rule. Technical specifications associated with ATWS modifications were submitted September 29, 1989, and are under staff review. A diversity issue associated with alternate rod injection and reactor pump trip analog trip units is under appeal to the NRC by the Boiling Water Reactors Owners Group.

USI A-46 Seismic Qualification of Equipment in Operating Plants

This issue was resolved with the issuance of GL 87-02, which endorsed the approach of using the seismic and test experience data proposed by the Seismic

Qualification Utility Group (SQUG) and Electric Power Research Institute (EPRI). This approach was endorsed by the Senior Seismic Review and Advisory Panel and approved by the NRC staff.

The scope of the review was narrowed to equipment required to bring each affected plant to hot shutdown and maintain it there for a minimum of 72 hours. The review includes a walkthrough of each plant which is required to inspect equipment. Evaluation of equipment will include: (a) adequacy of equipment anchorage; (b) functional capability of essential relays; (c) outliers and deficiencies (i.e., equipment with non-standard configurations); and (d) seismic systems interaction.

As an outgrowth of the Systematic Evaluation Program, the need was identified for reassessing design criteria and methods for the seismic qualification of mechanical equipment and electrical equipment. Therefore, the seismic qualification of the equipment in operating plants must be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The objective of this issue was to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at operating plants in lieu of attempting to backfit current design criteria for new plants.

Generic Letter 87-02, with associated guidance, required all affected utilities to evaluate the seismic adequacy of their plants. The specific requirements and approach for implementation are being developed jointly by SQUG and the staff on a generic basis before individual member utilities proceed with plant-specific implementation.

USI A-47 Safety Implications of Control Systems in LWR Nuclear Power Plants

In a letter dated March 23, 1990, the licensee provided a response to Generic Letter 89-19 related to USI A-47, Safety Implications of Control Systems. Two recommendations were contained in GL 89-19 related to BWRs: 1) that the reactor vessel overfill protection system should be separate from the control portion of the main feedwater control system so it is not powered from the same power source, not located in the same cabinet, and not routed so a fire is likely to affect both systems; and 2) that plant procedures and technical specifications in plants with main feedwater overfill protection include provisions to verify periodically the operability of these systems. Commonwealth Edison Company (CECo) in its response stated it is participating with the BWR Owners Group (BWROG) in the preparation of a generic response which will address the overfill protection/main feedwater (MFW) level control separation recommendations of the Generic Letter. In a letter dated April 2, 1990, the BWROG provided its generic response to these issues to the staff. CECo, in a June 8, 1990 letter, provided a plant specific response to the recommendations related to overfill protection and main feedwater (MFW) level control separation contained in the Generic Letter. CECo's conclusion was that the overfill protection configuration at Dresden is consistent with the "Group II" design identified in the Generic Letter and the "Group D" plant design identified in the BWROG response. However, CECo stated that its reactor pressure vessel overfill protection and MFW systems, although provided by separate power sources, does not conform to the physical separation recommendations contained in the Generic Letter in the following areas:

(1) The overfill protection system and the MFW control system have control switches, relays, and control modules located in the same cabinet.

(2) The field cables for the overfill protection system and the MFW control system have common routing points.

(3) The overfill protection system level sensors also provide input to the MFW control system. This input interfaces with the MFW control system runout flow control mode logic. The failure of a level sensor has the potential to disable the overfill protection system and the automatic reset of the runout flow control mode of the MFW control system. The impact on the MFW control system occurs during a high feedwater flow (run-out) condition. However, in the event of this condition, the operator still has the capability to manually reset the runout flow control mode and take actions to control level.

CECo estimated that it would cost in excess of $300,000 to implement the separation recommendations. The BWROG study indicated that the safety benefit gained by providing this separation was not cost effective. CECo has concurred with this conclusion and is not planning to initiate modifications to provide additional separation between the overfill protection system and MFW control system.

CECo, in its March 23, 1990 response also stated that at this time there are no technical specifications which address the reactor pressure vessel (RPV) overfill protection system. However, the reactor vessel water level instrumentation for overfill protection is calibrated every refuel outage. A logic system functional test is also performed every refuel outage. The calibration and functional tests are tracked by the station's general surveillance program. Additionally, the appropriate operating procedures will be revised to perform an instrument check of the overfill protection instrumentation on a daily basis. It is expected that the operating procedure revisions will be completed by August 1, 1990.

The acceptability of the licensee's responses to this Generic Letter are currently under staff review.

Conclusion

The licensee's schedule for response to these issues is dependent upon staff actions and is, therefore, acceptable.

6 CONCLUSIONS

On the basis of its evaluation of the application, the staff has determined that:

(1) The application for a full-term operating license (FTOL) for the Dresden Nuclear Power Station, Unit 2, filed by Commonwealth Edison Company on March 16, 1973, as supplemented and as revised, complies with the requirements of the Atomic Energy Act of 1954, as amended (Act), and the Commission's regulations as given in 10 CFR Chapter 1, except as duly exempted therefrom.

(2) Construction of the Dresden Nuclear Power Station, Unit 2, has been completed in conformity with Construction Permit CPPR-18, as amended, the application as amended, the provisions of the Act, and the rules and regulations of the Commission.

(3) The provisions of Provisional Operating License (POL) DPR-19 have been met.

(4) The facility will operate in conformity with the FTOL application as amended, the provisions of the Act, and the rules and regulations of the Commission.

(5) Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment (EA) and finding of no significant impact was prepared and published in the Federal Register on June 19, 1990 (SSR 24947). Accordingly, based upon the EA, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

(6) There is reasonable assurance (a) that the activities authorized by the FTOL can be conducted without endangering the health and safety of the public, and (b) that such activities will be conducted in compliance with regulations of the Commission set forth in 10 CFR Chapter 1.

(7) The licensee is technically qualified to engage in the activities authorized by the FTOL, in accordance with the regulations of the Commission set forth in 10 CFR Chapter 1.

(8) The issuance of the FTOL will not be inimical to the common defense and security or to the health and safety of the public.

(9) The FTOL for Dresden Nuclear Power Station, Unit 2, should be authorized by the NRC.

7 REFERENCES

Letters

October 21, 1969, from P. A. Morris (NRC), to W. E. Behnke (CECo), Subject: Committee's Review of Application for a License to Operate (Safety Evaluation, October 17, 1969), Dresden Unit 2

December 22, 1969, from P. A. Morris (NRC), to B. Lee, Jr. (CECo), Subject: Facility Received Provisional Operating License

November 11, 1972, from B. Lee, Jr. (CECo), to J. F. O'Leary (NRC), Subject: Application to Convert POL No. DPR-19 to a Full-Term License, AEC Docket No. 50-237

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July 16, 1982, from T. J. Rausch (CECo), to P. O'Connor (NRC), Subject: Dresden 2 SEP Topic III-1, Quality Group Classification of Components and Systems

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September 27, 1983, from B. Rybak (CECo), to H. R. Denton (NRC), Subject: Dresden Station Unit 2 Full Term Operating License

November 1, 1983, from D. G. Eisenhut (NRC), to All Boiling-Water Reactor Licensees, Subject: NUREG-0737 Technical Specifications--Generic Letter 83-36

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May 8, 1984, from D. G. Eisenhut (NRC), to All Licensees of Operating Reactors, Subject: Recombiner Capacity Requirements of 10 CFR 50.44(c)(3)(ii)--Generic Letter 84-09

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May 3, 1989, from G. M. Holahan (NRC), to C. Reed (CECo), Subject: 10 CFR 50.44, Active Combustible Gas Control System--Dresden and Quad Cities Nuclear Power Station 1 (TAC Nos. 56579, 56580, 55148, and 55149)

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October 30, 1989, from M. H. Richter (CECo) to Document Control Desk (NRC), Subject: Response to Generic Letter 89-16, NRC Docket Nos. 50-237/249

November 17, 1989, from J. A. Silady (CECo), to T. E. Murley (NRC), Subject: Dresden Nuclear Power Station Units 2 and 3 and Quad Cities Units 1 and 2, Request for Schedular Relief From DCRDR Annunciator Modifications, NRC Docket Nos. 50-237, 249, 254, and 265

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March 30, 1990, from J. A. Silady (CECo), to T. E. Murley (NRC), Subject: Dresden Nuclear Power Station Unit 2, Systematic Evaluation Program (SEP) Status, NRC Docket No. 50-237

February 21, 1990, from S. D. Floyd (BWROG), to F. J. Miraglia (NRC), Subject: Advises that Owners Group will develop generic design criteria for post-accident neutron monitoring to serve as a focal point for further discussion and review

March 12, 1990, from A. W. Dromerick (NRC), to E. E. Fitzpatrick (OCNGS), Subject: SEP Topic III-6, Seismic Design Considerations, Item 2 (TAC No. 49398)

March 23, 1990, from M. H. Richter (CECo) to USNRC Document Control Desk, Subject: Dresden Station Units 2 and 3 Response to Generic Letter 89-19, Docket Nos. 50-237/249

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June 15, 1990, from T. E. Murley (NRC) to T. J. Kovach (CECo), Subject: Staff's Backfit Analysis for Dresden Nuclear Power Station Units 2 and 3, Regarding Installation of a Hardened Wetwell Vent (Generic Letter 89-16)

<u>NRC Memoranda</u>

May 19, 1986, from G. C. Lainas to J. A. Zwolinski, D. R. Muller, and C. I. Grimes, Subject: Dresden Nuclear Plant, Units 2 and 3; Quad Cities Nuclear Plant, Units 1 and 2; Millstone Nuclear Plant 1; Cooper Nuclear Plant--Hydrogen Recombiner Capability (TAC Nos. 56579/56580, 55148/55149, and 55322)

November 9, 1989, from B. L. Siegel and T. M. Ross to J. W. Craig, Subject: Summary of Working Meeting with Commonwealth Edison Company to Discuss CECo's Station Blackout Submittal for the Dresden and Quad Cities Nuclear Plants

February 21, 1990, from B. L. Siegel to File, Subject: Status of Implementation of Unresolved Safety Issues for Dresden Nuclear Power Station Units 2-3

Other Reports

BWR Owners' Group, Letter from R. F. Janecek to T. E. Murley (NRC), Subject: BWR Owners' Group Licensing Topical Report, "Position on NRC Regulatory Guide 1.97, Revision 3 Requirements for Post-Accident Neutron Monitoring System" (General Electric Report NEDO-31558), April 1, 1988

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---, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," NUREG-0313, July 1977; Rev. 1, July 1980; Rev. 2, January 1988

---, "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants," NUREG-0531, February 1979

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---, "NRC Action Plan Developed as a Result of the TMI-2 Accident," NUREG-0600, Vols. 1 and 2, May 1980

---, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980

---, "Integrated Plant Safety Assessment, Systematic Evaluation Program, Dresden Nuclear Power Station Unit 2," NUREG-0823, February 1983; Supp. 1, October 1983

Commission finalized the Station Blackout Rule, 10 CFR 50.63, which resolves and supersedes Generic Task A-44 (53 FR 23203, June 21, 1988)

October 14, 1982, from R. C. DeYoung, Subject: IEB 82-03 SSINS NO. 6820 - Stress Corrosion Cracking in Thick-wall, Large-Diameter, Stainless Steel, Recirculation System Piping at BWR Plants

December 17, 1982, from D. G. Eisenhut (NRC), to All Licensees of Operating Reactors, Applicants for Operating Licensees, and Holders of Construction Permits, Subject: Supplement 1 to NUREG-0737--Requirements for Emergency Response Capability (Generic Letter 82-33)

January 17, 1983, SECY-83-19, from W. J. Dircks, to Commissioners, Subject: Conversion of Provisional Operating Licenses to Full Term Licenses

March 4, 1983, from R. C. DeYoung, Subject: IEB 83-02 SSINS NO. 6820 - Stress Corrosion Cracking in Large-Diameter, Stainless Steel Recirculation System Piping at BWR Plants

May 8, 1984, from D. G. Eisenhut (NRC), to All Licensees of Operating Reactors, Subject: Recombiner Capability Requirements of 10 CFR 50.44(c)(3)(ii) (Generic Letter No. 84-09)

October 26, 1984, from D. G. Eisenhut (NRC), to All Boiling Water Reactor (BWR) Licensees of Operating Reactors (except LaCrosse, Big Rock Point, Humbolt Bay and Dresden 1), Subject: Reactor Vessel Water Level Instrumentation in BWRs (Generic Letter No. 84-23)

April 19, 1984, from D. G. Eisenhut (NRC), to All Licensees of Operating Reactors, Applicants for Operating License, and Holders of Construction Permits for Boiling Water Reactors, Subject: Inspections of BWR Stainless Steel Piping (Generic Letter 84-11)

September 1, 1989, from J. G. Partlow (NRC), to All Holders of Operating Licenses for Nuclear Power Reactors with Mark I Containments, Subject: Installation of a Hardened Wetwell Vent (Generic Letter 89-16)

September 20, 1989, from J. G. Partlow (NRC) to All Licensees of Operating Reactors, Applicants for Operating Licenses and Holders of Construction Permits for Light Water Reactor Power Plants, Subject: Request for Action Related to the Resolution of Unresolved Safety Issue A-47 "Safety Implication of Control Systems in LWR Nuclear Power Plants" Pursuant to 10 CFR 50.54(f) (Generic Letter 89-19)

March 5, 1990, from B. L. Siegel (NRC) and L. N. Olshan (NRC), to J. W. Craig (NRC), Subject: Summary of Working Meeting With Commonwealth Edison Company (CECo) to Discuss CECo's Station Blackout Submittal for the Dresden and Quad Cities Nuclear Plants (TAC Nos. 68539, 68540, 68590 and 68591)

May 2, 1990, from L. N. Olshan (NRC) and P. L. Eng (NRC), to CECo, Subject: Meeting Summary - March 28, 1990 Meeting to Discuss Quad Cities/Dresden Approach to Station Blackout