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OFFICE OF INSPECTION AND ENFORCEMENT

Division of Quality Assurance, Vendor,
and Technical Training Center Programs

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Licensee: Commonwealth Edison Company
Post Office Box 767
Chicago, IL 60690

Facility Name: Dresden Station, Unit 3

Inspection At: Station Nuclear Engineering Department, Chicago, Illinois
Cygna Corporation, Chicago, Illinois
Nutech Engineers, Chicago, Illinois
Sargent & Lundy, Chicago, Illinois
Impell Corporation, Bannockburn, Illinois
Dresden Station, Morris, Illinois

Inspection Conducted: December 2-6, 16-20, 1985, and January 6-11, 23, 1986

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## LIST OF ABBREVIATIONS

| | |
|---|---|
| ACI | American Concrete Institute |
| AISC | American Institute of Steel Construction |
| ASME | American Society of Mechanical Engineers |
| ANSI | American National Standards Institute |
| ASTM | American Society for Testing and Materials |
| BWR | Boiling Water Reactor |
| CFR | Code of Federal Regulations |
| EQ | Equipment Qualification |
| ESF | Engineered Safety Features |
| FSAR | Final Safety Analysis Report |
| HELB | High Energy Line Break |
| HVAC | Heating, Ventilation and Air Conditioning |
| IEEE | Institute of Electrical and Electronics Engineers |
| LOCA | Loss of Coolant Accident |
| MR | Modification Request |
| NRC | U.S. Nuclear Regulatory Commission |
| NSSS | Nuclear Steam Supply System |
| P&ID | Piping and Instrumentation Diagram |
| RWCU | Reactor Water Cleanup |
| USAR | Updated Safety Analysis Report |
| CECo | Commonwealth Edison Company |
| SEP | Systematic Evaluation Program |
| SNED | Station Nuclear Engineering Department |

DRESDEN 3 SAFETY SYSTEM OUTAGE MODIFICATION INSPECTION (DESIGN)

# 1. INTRODUCTION AND SUMMARY

## 1.1 INTRODUCTION

This inspection was part of a trial NRC program for examining the adequacy of licensee management and control of modifications performed during major plant outages. The purpose of this portion of the Safety System Outage Modification Inspection Program was to examine, on a sampling basis, the detailed design and engineering required to support the outage.

### 1.1.1 REPORT FORMAT AND DEFINITIONS

Deficiencies, unresolved items, and observations are defined below and are included in an appendix to this report.

(1) Deficiencies

Errors, inconsistencies or procedure violations with regard to a specific licensing commitment, specification, procedure, code or regulation are described as deficiencies. Follow-up action is required for licensee resolution.

(2) Unresolved Items

Unresolved items are potential deficiencies which require more information to reach a conclusion. Follow-up action is required for licensee resolution.

(3) Observations

Observations represent cases where it is considered appropriate to call attention to matters that are not deficiencies or unresolved items. They include items recommended for licensee consideration, but for which, there is no specific regulatory requirement. No licensee response is required.

### 1.1.2 DRESDEN STATION ENGINEERING ORGANIZATION

Commonwealth Edison Company is the licensee for the Dresden Station. As such, Commonwealth Edison Company is responsible for the design, construction and operation of the facility. General Electric designed and provided the nuclear steam supply system. The original architect-engineer, Sargent and Lundy, is still under contract to the licensee. Other firms are also engaged to perform architect-engineering services for design modifications. These architect-engineers become responsible for the detailed preparation of the design basis specifications, drawings, instructions and procedures. As noted in CECo SNED Procedure A.3, SNED assumes a project management role for modifications at stations when outside

engineering firms are employed. This entails work scheduling, engineering manpower allocations, establishing the requisite level of engineering for specific areas (where required) and approving the format and content of engineering output. SNED is also responsible for design evaluation on all safety and ASME Code related modification work at operating stations. In addition, CECo SNED evaluates design basis documents in accordance with the criteria detailed in CECo Quality Assurance Manual Quality Procedure No. 3-1, Design Document Review and Control.

A set of controlled design documents for Dresden Units 2 and 3 are maintained in the Dresden Station records retention and retrieval center in accordance with CECo Quality Assurance Manual Quality Requirement No. 17.0, Quality Assurance Records. Each document is stored in hard copy, microfiche, and on cartridge and may be searched for either numerically or by key word.

### 1.1.3 INSPECTION EFFORT

The inspection was an interoffice NRC effort conducted with contractor assistance. Team members were selected to provide technical expertise and experience in the disciplines covered by the inspection. Most had previous experience as employees of architect-engineer firms or reactor manufacturers working on large commercial nuclear power plants. The others had related design experience on commercial nuclear facilities, test reactors, or naval reactors.

Beginning on September 3, 1985, a portion of the inspection team devoted one week to the initial study of background information and preparation of plans for the inspection. The majority of the team's inspection activities occurred at the Station Nuclear Engineering Offices, Chicago, Illinois the weeks of December 2 and 16, 1985, and January 6, 1986. The team also visited the engineering offices of various firms performing design and engineering work in support of the Dresden 3 outage. These included Cygna, Nutech, Impell, and Sargent and Lundy. The team also performed a plant tour of the Dresden Station Unit 3 and had discussions with site personnel. The inspection activities concluded on January 23, 1986 with an interim status briefing.

The inspection team reviewed the above organizations' staffing and procedures and interviewed personnel to determine the responsibilities of and the relationships among the entities involved in the design process. Primary emphasis was placed upon reviewing the adequacy of design details (or products) as a means of measuring how well the design process had functioned in the selected sampling area. In reviewing the design details, the team focused on the following items:

(1) Validity of design inputs and assumptions
(2) Validity of design specifications
(3) Validity of analyses
(4) Identification of system interface requirements
(5) Potential indirect effects of changes
(6) Proper component classification

(7) Revision control
(8) Application of design information transferred between organizations
(9) Design verification methods

The team inspected four engineering disciplines within the project: mechanical systems (Section 2), mechanical components (Section 3), instrumentation and controls (Section 4), and electric power (Section 5).

## 1.2 SUMMARY - MECHANICAL SYSTEMS

A leak detection system is being installed in the reactor water clean-up room in order to detect potential leakage from the high energy piping in sufficient time to enable this piping to be isolated from its source. The team was concerned that the leak detection system (which is non-safety-related) is being relied upon to protect safety-related equipment in the area, which has not been environmentally qualified for the effects of a high energy line break (i.e., water spray). In addition, the NRC has not approved the use of the leak-before-break concept in lieu of providing environmental qualification (Unresolved Item U2.1-3).

For a modification to the Units 1 and 2 diesel generators, there was no documented evidence to assure that original design requirements, including seismic qualification, had been imposed on the vendor (Deficiency D2.1-2). For the leak detection system, the design basis leakage flow rate and temperature were based on unconservative assumptions and undocumented input parameters (Deficiency D2.1-4). A modification to the high pressure coolant injection room cooler piping had inadequate justification to confirm the acceptability of pipe supports (Observation 2.2-1).

## 1.3 SUMMARY - MECHANICAL COMPONENTS

The Impell calculation which qualified temporary service reactor building penetrations did not consider the effects of pressure due to a high energy line break (HELB) (Deficiency D3.1-1).

The Sargent & Lundy calculation for the 125V battery rack did not adequately qualify the rack to the 33 Hz minimum frequency design criterion because it did not calculate the fundamental frequency of the base stringers which support the batteries (Observation 03.1-2). After this was pointed out by the team, Sargent & Lundy performed calculations which properly addressed the base stringers, and confirmed that the battery rack is seismically rigid.

Some instrument racks were not evaluated for thermal loads due to a HELB. The team questioned the validity of the response spectra generated at locations on the instrument rack mounting plates, since it was based on modeling in which instrument rack mounting plates were approximately modeled as a grid of beams. (Deficiency D3.1-3)

Impell had modified a spring hanger by replacing a pipe saddle with a U-bolt and a plate, but had not evaluated the plate for pipe support inertial forces caused by a design basis earthquake. This evaluation was necessary because the replacement plate had a lower load capacity than the pipe saddle, and an adjacent snubber had been removed. (Deficiency D3.1-5)

## 1.4 SUMMARY - INSTRUMENTATION & CONTROL

Approximately half of the reviewed design modifications were being accomplished in a controlled and technically acceptable manner. Five interrelated design modifications, involving the replacement of direct acting mechanical pressure and level sensors with analog electronic transmitter loops, contained a variety of technical design errors. In the most significant case, redundant reactor protection system transmitter trip channels were not separated in the analog trip system panels. (Deficiency D4.1-1) These errors violated the original plant design basis, as well as related FSAR and USAR licensing commitments and specific requirements stated in the modification approval letter. The analog transmitter modification packages involved six separate organizational units, and, in the team's opinion, at least one of these groups should have identified and resolved the technical design errors found in these modification packages.

The team identified several component classification errors in the Dresden 3 master equipment list as well as in one design input, and noted weaknesses in the design control process used to compile this list (Deficiencies D4.3-1 and D4.3-2). For the reactor water cleanup room leak detection system, analyses were not performed to confirm the acceptability of connecting non-Class 1E temperature monitor loads to Class 1E busses (Deficiency D4.4-1). A design basis requirement for an automatic rod block was missing from a rod worth minimizer upgrade modification without justification (Unresolved Item U4.5-1), and there was no indication of evaluation of the design basis for safety-related setpoint values. (Deficiency D4.2-1).

The individual items noted by the team appeared to be caused by several common factors, such as:

(1) Insufficient consideration of the original plant design basis. (Retrievability of design basis information appeared to be a significant obstacle);

(2) Inadequate review of architect-engineer design outputs;

(3) Inadequate indication of the technical thought process, i.e., no auditable document trail; and

(4) Inadequate design verification.

## 1.5 SUMMARY - ELECTRICAL POWER

The team reviewed 14 modifications concerning the dc distribution system, analog trip system, and the supporting design affected by the replacements of unqualified components with environmentally qualified components.

The team found errors in the design and testing of the dc systems, including improper determination of 250 V battery loads (Deficiency D5.1-1); improper sizing of 125 V battery cells (Deficiency D5.1-2); failure to update preliminary (unconservative) load current for a voltage drop calculation for a Unit 1 HPCI 125 V battery feed to Unit 2 (Deficiency D5.1-4); errors in design documentation (Deficiency D5.1-5); and acceptance criteria for battery surveillance temperatures that are not in agreement with the minimum battery temperatures in design basis calculations (Deficiency D5.1-8).

The team found no evidence that the cable voltage drop had been considered for the 24 V dc feeder cable to the analog trip system, even though the electronic trip system is sensitive to voltage variation (Deficiency D5.2-1).

The team found errors in the process of selecting and surveillance testing of motor operated valve overload relays (Deficiencies D5.3-1 and 5.3-3).

In general, the team found that the licensee failed to maintain an adequate internal design interface between its Station Nuclear Engineering Department and other Commonwealth Edison Company departments (e.g., Station Electrical Engineering and Dresden Operations/Maintenance) which led to incorrect design inputs and uncontrolled design outputs. The team also found a lack of external design interface control between SNED and the licensee's contracted engineering firms resulting in inadequate design inputs in calculations and specifications and incomplete design modification documentation.

## 1.6 OVERALL CONCLUSIONS

The results of the inspection at Dresden 3 revealed technical areas which indicate weaknesses in the design process associated with the modification packages reviewed. Thus, planned modifications not reviewed in detail by the team have the potential for similar weaknesses. We consider the following weaknesses particularly significant:

(1) The team questioned the appropriateness of using the leak-before-break concept as a basis for not providing required environmental qualification for safety-related equipment.

(2) Redundant reactor protection system transmitter trip channels were not separated in the analog trip system panels.

(3) The 125 volt battery cells were improperly sized and did not consider motor inrush currents.

In addition, the team identified programmatic weaknesses in the design process as follows:

(1) Modification design packages did not reference or make use of the original design bases for the plant feature being modified.

(2) There was inadequate coordination between SNED and other interfacing design organizations.

(3) Technical procedures did not exist to control the development of important engineering data, e.g., establishment of setpoints or the selection of dc overloads.

(4) Design control processes were not effective in detecting and resolving design errors.

With respect to items (2) and (4), the team found that even the potential "checks and balances" of having several organizations involved in the design process failed to identify design errors (e.g., analog trip system). On the other hand, multiorganization involvement sometimes contributed to potential or actual design errors by virtue of interfacing problems, (e.g., Deficiencies D2.1-4 and D4.3-1).

Although the report highlights problem areas and inspection concerns, the team also found positive aspects to the reviews conducted, which are not discussed in detail in the report. For example, mechanical interferences associated with the removal of recirculation piping were adequately documented.

## 2. MECHANICAL SYSTEMS

### 2.1 MECHANICAL DESIGN ACTIVITY TO SUPPORT CURRENT OUTAGE

A substantial effort during this outage involved the replacement of recirculation system piping susceptible to intergranular stress corrosion cracking (IGSCC). The team inspected two modification packages (M12-3-85-16 and M12-3-85-17) and supporting documentation involving the removal and reinstallation of mechanical interferences associated with replacement of the IGSCC piping. Impell is the architect-engineer responsible for the design work on these modifications. The team inspected a number of interference packages developed by Impell and the installer, Chicago Bridge and Iron, and found them to be adequately documented and generally consistent with Impell project instructions. Walkdowns were conducted to establish and document as-installed conditions, and the piping and other components were replaced in the original as-installed configuration. Such a reinstallation was designated as a "like-for-like" replacement. Based on the interference packages reviewed, the team found that the "like-for-like" designations were valid and represented no changes in design functions.

A number of non-safety-related reactor recirculation valve leak-off and valve body drain lines were permanently removed because they were no longer required. The team reviewed one interference package in which the removal of a valve leak-off line was designated a major change. The team found that

the modification approval letters did not indicate SNED approval of such design changes. The team was advised that the modification approval letters were to be revised to reflect authorization to remove these lines. However, the team considered that the modification approval letters should have been so revised when the change was first identified and prior to initiating related design work (Observation O2.1-1).

The emergency diesel generator diesel engine is equipped with a crankcase pressure detector to detect changes in normal negative pressure. The manufacturer of the diesel generator found that the baffle previously installed between the diesel generator engine crankcase pressure detector and the engine accessory gear train housing was not effective in all cases, and developed a new baffle design. The team inspected Modification M12-3-84-40 which involved installation of the new baffle plate on the safety-related diesel generators, and found no indication that the new design met the original design requirements, including seismic (Deficiency D2.1-2).

The lines supplying heat exchangers in the reactor water clean-up system heat exchanger room are high energy lines which can be isolated in the event of postulated high energy line breaks. A non-safety-related leak detection system (to be installed in the reactor water clean-up room under Modification M12-2(3)-84-17) is intended to alert the operator to a leak to permit isolation of the line prior to it developing into a high energy line break. None of the equipment affected by a postulated high energy line break was qualified for this environment. Since the leak detection system is non-safety-related (no credit can be taken for its function), safety-related equipment could be compromised by the harsh environment resulting from the undetected leak and the ensuing high energy line break. The team considers these issues unresolved subject to further review by the Staff as to the appropriateness of not qualifying the equipment for a harsh environment (Unresolved Item U2.1-3).

The reactor water clean-up room leak detection system consists of five resistance temperature detectors in each of the dual trains, which will alarm in the control room. The team reviewed the basis for the 150°F leak detection setting and found inconsistent data for the leakage flow upon which this setting is based. The temperature transient calculations included assumptions which maximized temperatures in the room resulting from the leakage, which are unconservative. There was no indication of the basis for the initial conditions of temperature, pressure and humidity assumed in the calculations. There was no consideration of the effects of low ambient temperatures, ventilation, and drainage of the leakage accumulated in the room, all of which would lower the temperature. Consequently, it was not clear that the design basis leakage was either defined or detectable by the leak detection system to be installed (Deficiency D2.1-4). The team found that these issues reflected upon inadequate interfaces between SNED, Bechtel and Nutech, including Nutech's inadequate review of design input from Bechtel.

## 2.2 DESIGN ACTIVITY ASSOCIATED WITH COMPLETED MODIFICATIONS

Modification M12-3-84-14 involves installation of a four way valve in the piping to the containment cooling service water pump vault coolers during the current outage in order to permit flow reversal through the cooler. Since the design work for this effort is not yet complete, the team reviewed a modification completed during a previous outage (M12-3-81-31) which was similar to the latter modification (for both, the design work was performed by Sargent & Lundy). It provided for the addition of valves and seismic Class 1 piping to permit flow reversal through the high pressure coolant injection room cooler. The field had been advised to increase the 1/2" rod size on the existing pipe support to 3/4" to sustain any increased loads due to the modification. The team found no analysis to confirm the acceptability of the revised configuration (Observation 2.2-1).

# 3. MECHANICAL COMPONENTS

The team evaluated five modifications to piping systems and equipment and several equipment supports identified during a plant tour. The team evaluated the related piping, pipe supports and equipment for conformance to the Dresden FSAR and governing codes, standards and specifications.

## 3.1 PIPING/EQUIPMENT REVIEW

Modification M12-3-85-53 provides for the installation of twelve five-inch diameter penetrations in the Unit 3 reactor building exterior wall to allow services such as gas, electrical and welding leads to be run into the reactor building during the replacement of stainless steel recirculation pipe subject to intergranular stress corrosion cracking (IGSCC). The Impell calculation which qualified the reactor building penetrations did not consider the effects of pressure due to a high energy line break (HELB) (Deficiency D3.1-1).

Modification M12-3-85-25 replaced the Dresden Unit 3 125 volt battery with one of larger capacity, and the existing battery rack with a new seismically qualified battery rack. Since the Sargent & Lundy design for the new Dresden 3 battery rack was not complete at the time of the inspection, the team reviewed the Sargent & Lundy design for the Dresden 2 125 volt battery rack. The structural calculation did not adequately qualify the battery rack to the 33 Hz minimum frequency design criterion for ensuring that the rack is seismically rigid. The fundamental frequency of the base stringers which support the batteries was not calculated because the battery masses were lumped at the joints of the battery rack analytical model. To address this issue after it was pointed out by the team, Sargent & Lundy prepared an addendum to the original calculation which lumps the battery masses to the support stringers. The base stringer fundamental frequency calculated for this revised analysis is 26.3 Hz. However, an examination of the design response spectra for Dresden Units 2 and 3 indicates that zero period acceleration (ZPA) values govern for frequencies greater than approximately 25 Hz. Therefore, the battery rack can be considered seismically rigid, and therefore complies with the intent of the above design criterion. (Observation 03.1-2).

Modification M12-2(3)-83-40 replaces existing instruments with new environmentally and seismically qualified instruments. The team reviewed the seismic qualification of the instrument racks on which the new instruments have been mounted, and the generation of required response spectra at these instrument mounting locations. In-situ testing of instrument racks had been inconclusive due to the system non-linearities caused by rattling of instruments, loose pipe and conduit, and local "noise" at mounting plate locations. The team recommends that Section 2.2 of NUREG-1030, Development and Assessment of In-Situ Testing Methods to Assist in Qualification of Equipment, be reviewed by CECo with respect to these inconclusive findings.

Several instrument racks appear to be subject to relatively high temperatures due to HELB. However, the instrument racks were not qualified for this thermal load. The Cygna in-house computer program used to perform the instrument rack seismic analysis cannot model two-dimensional (plate) elements. Therefore, the instrument rack mounting plates are approximately modeled as a grid of beams. The team questioned the validity of the approximation, noting that inaccurate modeling of the mounting plates would invalidate the response spectra generated at locations on the mounting plates (Deficiency D3.1-3).

Some modification packages did not reference the Sargent & Lundy and General Electric design specifications which governed the original design and procurement of NSSS and BOP piping systems and equipment. In order to ensure that design work associated with modification packages is consistent with the original design basis, it is important to review the design specifications for the original design when evaluating the modification package. To ensure that this process occurs, it is prudent for the modification package to reference the original design specifications. The team recommends that CECO procedures be revised to make this a requirement (Observation O3.1-4).

Modification M12-3-85-16 authorizes the replacement of critical piping susceptible to intergranular stress corrosion cracking (IGSCC) at Dresden Unit 3. The team reviewed the Impell specifications and procedures for this modification and the modified design of two pipe supports, and found that a spring hanger had been modified by replacing a pipe saddle with a U-bolt and a plate. However, the plate had not been evaluated for pipe support inertial forces caused by a design basis earthquake. This evaluation was necessary because the replacement plate had a lower load capacity than the pipe saddle, and an adjacent snubber present in the original piping configuration had been deleted (Deficiency D3.1-5).

The Sargent & Lundy mechanical and electrical equipment list references the design specifications used to procure the original battery chargers for Dresden Unit 3. Revised specifications subsequently used to purchase replacement equipment were not referenced (Observation O3.1-6).

## 4. INSTRUMENTATION AND CONTROL

The team reviewed design modification packages for: (1) consistency with Dresden 3 design basis requirements, (2) conformance with applicable

regulatory criteria and FSAR/USAR commitments, (3) technical adequacy of the design approach, and (4) completeness of design details and independent verification reviews.

The team reviewed the following sixteen instrumentation and control design modification packages:

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|---|---|
| M12-2-74-16 | Torus to Drywell Vacuum Breaker Switch |
| M12-2-74-143 | Core Spray Flow Switch |
| M12-3-81-04 | Scram Valve Control Low Pressure Air Dump |
| M12-3-83-36 | Core Spray Sys. Flow Transmitters (Analog Trip) |
| M12-3-83-37 | Pressure Suppression Transmitters (Analog Trip) |
| M12-3-83-38 | HPCI Pressure and DP Transmitters (Analog Trip) |
| M12-3-83-39 | LPCI Flow and DP Transmitters (Analog Trip) |
| M12-3-83-40 | Reactor Vessel Level Transmitters (Analog Trip) |
| M12-3-83-56 | Feedwater Control System Upgrade |
| M12-3-83-58 | Automatic Depressurization System Logic |
| M12-3-84-04 | Rod Worth Minimizer Computer and Output Buffer |
| M12-3-84-06 | Rod Position Indication System Output Buffer |
| M12-3-84-09 | Core Spray Valve Hammering |
| M12-3-84-17 | Reactor Water Cleanup Room Leak Detection |
| M12-3-84-118 | Post-Scram Drywell Pressurization |
| M12-3-84-139 | Drywell Water Level Indication |

The three pre-1983 design modification packages were reviewed with Dresden site personnel for safety-related setpoint determinations. Because of their preliminary design status, the team reviewed only the currently planned design approach for modifications 83-56, 83-58, 84-118, and 84-139. The team had no questions regarding modification 84-09.

The remaining eight design modification packages for the analog trip system transmitter upgrade, reactor water cleanup room leak detection, and rod worth minimizer upgrade were reviewed in depth with Commonwealth Edison and Nutech engineers. These modifications contained design errors relative to the original design basis for Dresden 3, FSAR or USAR commitments, or Commonwealth Edison modification approval letter requirements. For the five analog trip system design modifications, the design deficiencies identified by the team indicate that design control processes used by Station Nuclear Engineering, the technical staffs at Dresden and Quad-Cities Stations, Bechtel, Nutherm, and Nutech have not been effective in detecting and resolving individual design errors.

## 4.1 ANALOG TRIP SYSTEM TRANSMITTER DESIGN MODIFICATION

In the 83-36 through 83-40 design modifications for the analog trip system, electronic transmitters replaced direct acting pressure and level switches. These modifications affected portions of the reactor protection, containment isolation, and several engineered safety feature systems. Redundant reactor protection system transmitter trip channels that require physical separation and electrical isolation to satisfy the independence criteria of IEEE Proposed Standard 279-1968 were not separated in analog trip system panels as required by General Electric Separation Specification 22A2501. Explicit separation guidance provided in General Electric Licensing Topical Report NEDO-21617A, which was referenced in the Commonwealth Edison modification approval letter, was not used for this design change (Deficiency D4.1-1).

Based on the team's discussions with Nutech the initial design modification met the separation criteria, but because it did not conform with the power source assignments of the original reactor protection system design, the probability of spurious reactor trip was significantly increased. The subsequently revised design corrected the spurious reactor trip issue; however, it introduced a channel separation violation within each of the analog trip system cabinets.

As a consequence of this separation violation, a single event within an analog trip system cabinet could disable the reactor vessel water level trips of the reactor protection and primary containment isolation systems. Reactor vessel water level does not have a functionally diverse backup variable that would be effective for postulated transient and accident design basis events.

Separation of non-safety-related wiring from safety-related wiring within the analog trip system panels was required (by a note on the Nutech drawings) to be indicated on drawings. However, the drawings reviewed by the team did not identify most of the non-safety wiring requiring separation (Deficiency D4.1-1).

Although the Commonwealth Edison modification approval letter stated that the intended design "changes are consistent with General Electric Licensing Topical Report NEDO-21617A," the design modification differed from the General Electric report in its use of two cabinets (versus four) as well as in the coordination of process sensor physical connections with the safety system coincidence logic. Hence, the applicability of various General Electric analyses (contained in the report) to the Dresden 3 Station could not be confirmed (Deficiency D4.1-1).

There was no evidence that the original reliability requirements were satisfied for the transmitter channel replacement components, even though the FSAR and the General Electric Licensing Topical Report provided minimum reliability quantitative values (Deficiency D4.1-2).

There was no evidence that failure mode and effects analyses were performed for the transmitter channels to account for additional internal failure modes introduced by the modification. The modification approval letter required design compliance to IEEE Std. 352-1975 which provides failure mode and effects analysis methodology (Deficiency D4.1-2).

The General Electric Licensing Topical Report indicated that transmitter response times were longer than those for the replaced direct acting mechanical switches, but this was not assessed relative to original assumptions made in the FSAR transient and accident analyses (Deficiency D4.1-2).

Two interruptible Class 1E power sources replaced the high reliability reactor protection system motor-generator and a 125 volt dc uninterruptible power source used in the original Dresden 3 design. This change differed from the original design basis for the affected circuits and with respect to vital instrumentation power supply criteria described in IEEE Std. 308-1971.

With Nutech assistance, the team reviewed those transmitter substitutions affected by this change and determined that a postulated power outage interval of approximately 10 seconds between the loss of offsite power and operation of the emergency diesel-generators would place the reactor protection and containment isolation systems in a safe "half-scram" and "half-isolation" condition respectively. Control room operator manual reset actions would be required to restore these systems to their normal operating status. Manual reset action is not required for the engineered safety feature circuits; however, some of these ESF sensor channels would now not be operable until ac power has been restored by the diesel-generators. The impact of this change on emergency core cooling system availability, such as described for a large loss of coolant accident in USAR Table 6.2.7.4, was not assessed (Unresolved Item U4.1-3).

With these design modifications, high and low pressure emergency core cooling systems have been made dependent upon the the 24 volt dc neutron monitoring system batteries. Systems interaction effects as well as individual effects on the high pressure coolant injection and low pressure coolant injection systems performance were not assessed (Unresolved Item U4.1-4).

Environmental qualification documentation for the analog trip system transmitter and instrument loop components contained a number of inconsistencies (Unresolved Item U4.1-5).

A common instrument line for two redundant torus pressure switch instruments was present in the original balance of plant design, which is a violation of the single failure criterion stated in IEEE Proposed Standard 279-1968. The team considers that this violation should have been identified in the design work for the transmitter replacement (Unresolved Item U4.1-6).

The team reviewed the adequacy of safety evaluations performed pursuant to 10CFR50.59 and found that, for Modifications M12-3-83-36 through -40 involving the analog trip system, there was no evaluation of the impact of changes in system response time, power supply interfaces, and channel separation (Deficiency D4.1-7).

## 4.2 SETPOINT CALCULATIONS

The team reviewed analog transmitter instrument loops for the determinations of setpoint value, accuracy, and tolerance relative to instrument range. There was no evaluation of the design basis for setpoint values pertinent to the transmitter substitution modifications as required by Section 3 of IEEE Proposed Standard 279-1968. The team was unable to locate individual setpoint calculations during the inspection (Deficiency D4.2-1).

## 4.3 MASTER EQUIPMENT LIST

Sargent & Lundy compiled a Dresden 3 master equipment list using input provided by Bechtel and Nutech as well as data from their own records. Nutech's input list incorrectly designated a number of safety-related

reactor protection system, high pressure coolant injection system, and low pressure coolant injection components as non-safety-related. Nutech provided two different versions of this list to Commonwealth Edison, but identified each as Revision 1 even though the technical content was not identical. In addition, Nutech did not review and verify the design input in accordance with its internal procedures. Sargent and Lundy compiled the master equipment list on the basis that each identified component was safety-related without any apparent consideration of the Nutech input (Deficiency D4.3-1).

For the above deficiency, although the design control process among interfacing organizations was questionable, the master equipment list was correct. However, in other cases, the master equipment list incorrectly designated safety-related motor and motor control circuit components as non-safety-related, such as a HPCI turbine steam supply valve, an HPCI suction supply valve, HPCI return line valves to the condensate storage tank, and containment cooling service water pump motors (Deficiency D4.3-2).

### 4.4 REACTOR WATER CLEANUP ROOM LEAK DETECTION

Two non-safety-related temperature monitors have been added to detect leaks in ASME class 2 piping in the rooms containing reactor water cleanup system equipment. For this design modification, non-Class 1E panels, 2203-77A and B, are supplied power from two Class 1E busses, 38-1 and 39-1.

Non-Class 1E loads were not adequately isolated from these Class 1E sources, such as by the use of double circuit breakers. For such cases, Commonwealth Edison had committed to provide a short circuit breaker coordination study to verify that the faults in the non-Class 1E portion will not affect the Class 1E busses. The breaker coordination study had not been performed prior to the inspection (Deficiency D4.4-1). After the team identified the omission, it has been subsequently demonstrated that Class 1E busses will not be degraded below an acceptable level.

### 4.5 ROD WORTH MINIMIZER UPGRADE

Two design modification packages for upgrading of the rod worth minimizer were reviewed by the team. In the original design, when the rod worth minimizer equipment was inoperative, there was an automatic rod movement block. The USAR specifically addresses this design feature. In the modification, an operator alarm was substituted for this automatic rod block; however, this change was not justified or evaluated (Unresolved Item U4.5-1).

## 5. ELECTRICAL POWER

The primary electrical power modifications planned for this outage were replacement of non-qualified electrical components with qualified components and modifications to the dc system. The modification packages for the EQ replacements were complete and those for the batteries were still under review. A previously installed modification relating to current modifications was also reviewed.

## 5.1 DC SYSTEM MODIFICATIONS

The team reviewed the modification packages being prepared for the replacement of aging and undersized 125 volt and 250 volt batteries. Sargent & Lundy is preparing the safety-related calculations required as the sizing basis for the batteries. The 125 V battery sizing calculation departed from industry practice and selected a smaller cell size than the calculation suggested (Deficiency D5.1-2). Sargent & Lundy did not consider the motor inrush currents or use correct temperature correction factors in sizing the 250 V battery (Deficiency D5.1-1). The team noted that existing batteries are required to be maintained at or above 60°F; however, these batteries were sized for a 77°F ambient. The 17°F temperature difference results in an 11% loss of capacity at 60°F (Deficiency D5.1-8).

A modification to provide a temporary feed to the Unit 2 125 V dc bus from the Unit 1 HPCI battery (until the Unit 2 battery could be replaced) was based on a voltage drop calculation which used preliminary load data. There was no evaluation of the need to verify this preliminary load data, which is less conservative than the load data used to size the replacement 125 V batteries (Deficiency D5.1-4).

The original 24/48 volt batteries, rated 80 ampere-hours, were replaced with batteries rated 190 ampere-hours. However, the key 24/48 volt single line drawing and the Updated Safety Analysis Report were not correctly updated to reflect the change in ampere-hour rating (Deficiency D5.1-5).

## 5.2 ANALOG TRIP SYSTEM

An electronic system, which incorporated relay contact outputs, replaced process actuated switches. During the design phase, it was decided to power part of the electronic system from the unregulated 24/48 volt dc distribution system. There was no evidence of the method of the feeder cable size selection, even though the equipment is sensitive to low voltage operation and the cable voltage drop is a limiting condition for sizing the dc power supply cable (Deficiency D5.2-1). Nutech and Bechtel were aware of the fact that the Agastat relays were not rated for operation at the battery equalizing voltage of 28 volts and reviewed the effects of operation at this voltage with the manufacturer. The conclusion of these discussions was that intermittent high voltage operation (0.3 volts above the relay rating) would not affect relay performance (Observation O5.2-2).

## 5.3 MOTOR REPLACEMENT

Non-qualified motors on motor operated valves (MOVs) are being replaced with qualified motors. The selection of motor overload protection is performed by the Station Electrical Engineering Department using a draft procedure for selecting the overloads. It appears that Station Electrical has reviewed

only the ac MOV overload relays requested by Station Nuclear. There is no procedure for selecting dc overload relays, and it appears that the dc overloads have never been reviewed by Station Electrical. The overload selection procedure does not address the setpoint of the ±15% adjustment dial on the GE124K relays, nor is there evidence of Station Electrical making any recommendations for the setpoint (Deficiency D5.3-3).

The test procedure for periodically verifying calibration of MOV overload relays (as submitted to the NRC in response to a Systematic Evaluation Program concern) was reviewed by the team. The original acceptance limits were based on a GE curve used to size overloads. The acceptance limits were increased based on a second GE curve and field test data, both of which were outside the original acceptance range. Since the GE curve used to size overloads is no longer the basis for the acceptance limits, the overloads may not provide required protection. The procedure fails to list all the 480 volt ac safety-related motor operated valves listed in CECo Procedure DMP-04-G. There is no test procedure for 208 volt ac or dc safety-related valves listed in Procedure DMP-040-G (Deficiency D5.3-3).

The non-qualified LPCI room cooler fan motors are being replaced with environmentally qualified motors of the same rating on a "one-for-one" basis. There was no evidence indicating compliance with the specification requirements for the bearing life (9 year L10 bearing life) and the motor's ability to deliver full load torque and horsepower without damage at 75% of rated voltage for one minute intervals (Deficiency D5.3-2).

## 6.0 PERSONS CONTACTED

### MECHANICAL SYSTEMS

| Name | Title | Organization |
|---|---|---|
| B. Fancher | Engineer | SNED |
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| J. Hausemen | Project Engineer | SNED |
| Z. Boxer | Engineer | SNED |
| D. Wilgis | Engineer | SNED |
| M. Frederick | Engineer | SNED |
| M. Jackson | Engineer | SNED |
| R. Mirochna | Engineer | SNED |
| R. Odegard | Quality Assurance Engineer | S&L |
| S. Taylor | Manager | S&L |
| G. Jurkin | Mechanical Engineer | S&L |
| R. May | Lead Engineer | Impell |
| T. Miller | Assistant Project Manager | Impell |
| T. Mc Kinney | Division Manager | Impell |
| T. Wittig | Division Manager | Impell |
| J. Famiglietti | Project Manager | Impell |
| N. Lane | Electrical Engineer | Impell |
| T. Little | Engineer | Impell |
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| D.J. Scott | Station Manager | Station Engineering |
| R. Dyer | Engineer | Station Engineering |
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### MECHANICAL COMPONENTS

| Name | Title | Organization |
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| M.S. Tucker | Principal Engineer (SNED) | CECo |
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| Z. Boxer | Engineer (SNED) | CECo |
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| J. Brunner | Assistant Superintendent Technical Services (Dresden) | CECo |
| R. Flessner | Services Superintendent (Dresden) | CECo |
| D.J. Scott | Station Manager (Dresden) | CECo |
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MECHANICAL COMPONENTS (Continued)

| Name | Title | Organization |
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INSTRUMENTATION AND CONTROL

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| R.M. Crawford | Executive Director | Nutech |
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| R.N. Beba | Instrum. & Control | Nutech |
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ELECTRICAL

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## LIST OF DEFICIENCIES, UNRESOLVED ITEMS AND OBSERVATIONS

| Item | | Title |
|---|---|---|
| O2.1-1 | (Observation) | Removal of Valve Leak-Off Lines |
| D2.1-2 | (Deficiency) | Baffle Plate for Emergency Diesel Generator |
| U2.1-3 | (Unresolved) | Reactor Water Clean-Up System Leak Detection |
| D2.1-4 | (Deficiency) | Reactor Water Clean-Up Leak Detection System Parameters |
| O2.2-1 | (Observation) | HPCI Room Cooler Modification |
| D3.1-1 | (Deficiency) | Reactor Building Service Penetrations |
| O3.1-2 | (Observation) | 125 Volt Battery Rack Seismic Qualification |
| D3.1-3 | (Deficiency) | Instrument Rack Modifications |
| O3.1-4 | (Observation) | Failure to Reference Original Design Specifications |
| D3.1-5 | (Deficiency) | IGSCC Pipe Removal and Replacement |
| O3.1-6 | (Observation) | Battery Charger Specifications |
| D4.1-1 | (Deficiency) | Separation of Redundant Class 1E Channels and Separation of Wiring for Safety-Related and Non-Safety-Related Circuits for Analog Transmitter Replacement Modifications |
| D4.1-2 | (Deficiency) | Reliability, Failure Modes, and Response Time Requirements for Analog Transmitter Replacement Modifications |
| U4.1-3 | (Unresolved) | Selection of Interruptible Power Sources for Analog Transmitter Replacement Modifications |

## LIST OF DEFICIENCIES, UNRESOLVED ITEMS AND OBSERVATIONS

| Item | | Title |
|---|---|---|
| U4.1-4 | (Unresolved) | High and Low Pressure Coolant Injection System Dependency on Neutron Monitoring System Battery Sources for Analog Transmitter Replacement Modifications |
| U4.1-5 | (Unresolved) | Environmental Qualification Documentation for Analog Transmitter Replacement Modifications |
| U4.1-6 | (Unresolved) | Common Instrument Line to Redundant Instruments for Analog Transmitter Replacement Modifications |
| D4.1-7 | (Deficiency) | Failure to Adequately Document Safety Evaluations for Modifications |
| D4.2-1 | (Deficiency) | Setpoint Calculations for Safety-Related Instruments |
| D4.3-1 | (Deficiency) | Component Classification Errors |
| D4.3-2 | (Deficiency) | Safety Classification of Master Equipment List Motor Components |
| D4.4-1 | (Deficiency) | Reactor Water Clean-Up Room Leak Detection System Safety-Related and Non-Safety-Related Circuit Isolation |
| U4.5-1 | (Unresolved) | Rod Worth Minimizer Design Basis Requirements |
| D5.1-1 | (Deficiency) | 250 V Battery Sizing Calculation |
| D5.1-2 | (Deficiency) | 125 V Battery Cell Selection |
| O5.1-3 | (Observation) | 125 V Battery Sizing Calculation |
| D5.1-4 | (Deficiency) | Unit 1 HPCI Battery Tie Voltage Drop Calculation |
| D5.1-5 | (Deficiency) | Errors in Updated Documentation for 24/48 Volt Batteries |

LIST OF DEFICIENCIES, UNRESOLVED ITEMS AND OBSERVATIONS

| Item | | Title |
|---|---|---|
| O5.1-6 | (Observation) | Specification of 24/48 Volt Batteries |
| O5.1-7 | (Observation) | DC Breaker Coordination |
| D5.1-8 | (Deficiency) | Battery Temperature Surveillance for Existing Battery |
| D5.2-1 | (Deficiency) | Documentation Lacking for Cable Selection |
| O5.2-2 | (Observation) | Maximum Voltage for Agastat Relays |
| D5.3-1 | (Deficiency) | Motor Operated Valve Protection |
| D5.3-2 | (Deficiency) | Review of LPCI Room Cooler Motor Replacement |
| D5.3-3 | (Deficiency) | MOV Contactor Test Procedure |

02.1-1 (Observation) Removal of Valve Leak-off Lines

DESCRIPTION: The reactor recirculation valve leak-off lines provide a leakage path from the valve stuffing boxes to the equipment drain sump. Reactor recirculation valve body drain lines are provided to drain the piping upstream and downstream of the valves. A number of these lines are being permanently removed from the reactor recirculation system piping in the process of removing mechanical interferences associated with the replacement of piping susceptible to intergranular stress corrosion cracking (IGSCC).

The team could find no evidence that the modification approval letters (References 3 and 4) had been revised to reflect removal of these leak-off/drain lines. SNED intends to issue a revised modification approval letter to reflect these design changes prior to completion of the modification. Although Commonwealth procedures indicate a requirement for the issue of a modification approval letter by SNED (paragraph C 6.e. of Reference 7) there is no specific requirement for the revision of the modification approval letter when a change in scope is identified. The team recommends that revision to the letter be required in such cases to ensure that out-of-scope design changes are not initiated within a modification without invoking the same SNED review and approval process as for the original modification design.

REFERENCES


1. Impell Corporation Project Instruction PI-8, Control, Review and Distribution of Mechanical/Structural Interference Packages, Revision 2, September 27, 1985.
2. Chicago Bridge and Iron Interference Identification Sheet, Tag 125, Revision 1, June 24, 1985.
3. SNED Modification Approval Letter, "Reactor Recirculation System Pipe Removal and Replacement," M12-3-85-16, September 10, 1985.
4. SNED Modification Approval Letters, "Mechanical Interference Removal and Reinstallation," M12-3-85-17, June 21 and August 27, 1985.
5. Impell Corporation Letter to SNED, "Dresden Unit 3 Pipe Replacement Project, Deletion of Large Bore Valve Leak-off and Drain Lines," 0590-010-355, March 18, 1985.
6. Dresden Station Letter to SNED, "Dresden Unit 3 Pipe Replacement Project, Deletion of Valve Leak-off and Drain Lines," DJS LTR #85-825, August 14, 1985.
7. Commonwealth Edison Company Quality Procedure Q.P. No. 3-51, Design Control for Operations - Plant Modifications, Revision 12, March 13, 1985.
8. Impell Project Instruction PI-9, 10 CFR 50.59 Evaluations, Revision 1, October 9,1985.
9. Impell Corporation Letter to SNED, "Safety Evaluation Transmittal," 0595-010-1659, September 17, 1985.

D2.1-2 (Deficiency) Baffle Plate for Emergency Diesel Generator

DESCRIPTION: The diesel engine is equipped with a crankcase pressure detector. Low pressure will initiate an engine shutdown. The team reviewed a modification (applicable to diesel generators for both Units 1 and 2) to the design of a baffle plate between the pressure detector and engine accessory gear train housing.

In a letter (Reference 1) to SNED, General Motors, Electro-Motive Division (EMD), the manufacturer of the diesel generator, indicated that the baffle, which had been previously installed to prevent false crankcase pressure alarms on fast starts, was not effective in all cases. The letter stated that a new baffle design was developed and proven effective in shop and field testing. The SNED modification approval letter (Reference 2) states, "the DG were built to Sargent and Lundy Specification K-2183, dated 5/24/66. Codes are not applicable to this modification. Revision to existing stress reports will not be required." The team found no documentation to substantiate that the modified baffle plate had been evaluated to assure its design is consistent with the original seismic and other design requirements for the diesel generator. Station Engineering purchased the modified baffle from EMD; however, there was no indication that the purchase order imposed original design requirements on the vendor. SNED informed the team that a formal request is to be made to EMD to confirm that the baffle design is consistent with the original seismic and other code requirements.

BASIS: The licensee has committed to Regulatory Guide 1.64 (References 3 and 4) which endorses ANSI N45.2.11 (Reference 5). This requires the plant design to be traceable to the original design input. Contrary to this requirement, no evidence exists that the design of the planned modification (baffle plate) to the diesel generator has been analyzed to assure that it is consistent with original design/seismic requirements.

REFERENCES

1. Electro-Motive Division General Motors Corporation Letter to SNED, August 30, 1984.
2. SNED Modification Approval Letter, "Installing Baffles Between DG Engine Crankcase Detector and Engine Accessory Gear Train Housing," Modification No. M12-2(3)-84-40, September 17, 1984.
3. Commonwealth Edison Company Quality Assurance Manual, Quality Requirement 2.0.
4. Regulatory Guide 1.64, Quality Assurance Requirements for the Design of Nuclear Power Plants, 1974
5. ANSI N45.2.11, Quality Assurance Requirements for the Design of Nuclear Power Plants, 1974.

U2.1-3 (Unresolved Item) Reactor Water Clean-up System Leak Detection

DESCRIPTION: The lines supplying heat exchangers (and auxiliary pump) in the reactor water clean-up system heat exchanger room are high energy lines and are therefore subject to the potential for breaks which might produce harsh environments. Flow to this equipment can be terminated using safety-related valves 3-1201-2 and 3-1201-3 which are located in the same general area. Modification M12-2(3)-84-17 covers installation of a non-safety-related leak detection system in the reactor water clean-up room as part of the Dresden equipment environmental qualification program and IE Bulletin 79-01B. The system consists of five resistance temperature detectors in each of the dual trains which will alarm in the control room. The intent is to alert the operator to the leak in sufficient time to isolate the line and prevent a high energy line break which could compromise safety-related equipment affected by the resulting harsh environment.

Postulation of a high energy line break is a design basis event which should be addressed as to environmental effects on safety-related equipment in accordance with the requirements of 10 CFR 50.49. SNED informed the team that the NRC had accepted the Commonwealth position concerning the use of the leak-before-break concept as a basis for not providing required environmental qualification for safety-related equipment. The team could find no documented evidence to indicate acceptance by the NRC that the environmental effects of a postulated high energy line break on safety-related equipment could be avoided by detecting the leak prior to the development of a full line break. The Systematic Evaluation Program Final Report (Reference 2) for Dresden 2, which is applicable to Dresden 3, indicates that the licensee's approach utilizing the leak-before-break concept was found to be not acceptable (Reference 2, Section 4.7.3) except in the areas of pipe whip load formulation and the interaction of pipe whip and jet impingement with the containment liner. Section 4.13 of Reference 2 discusses studies conducted concerning leak detection as it related to the small break LOCA. However, the report (SEP) stated, these studies did 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)...." No documented evidence was presented to the team to indicate that this position had been changed.

The team reviewed a Nutech report (Reference 3) entitled "Leak Before Break Justification, Reactor Water Cleanup System, Dresden Station Units 2 & 3, and Quad Cities Station Units 1 & 2." This report is a summary of stress analyses to substantiate the leak-before-break concept and its application to the reactor water clean-up system piping. Based on the results of these analyses, the report concludes that "...the proposed leak detection system is adequate because catastrophic pipe failure is not expected and the temperature monitors will alert the reactor operator(s) to leaking pipes long before the leaks attain critical size." The team found that the report did not contain sufficient detail to permit verification of this conclusion.

SNED informed the team that the leak detection system is designed to be non-safety-related based on the rationale that it is an anticipatory backup system and does not meet 10CFR50.49 (b)(1) criteria for safety-related equipment. Even assuming that the leak-before-break concept could be demonstrated to be an acceptable means of preventing the development of a full line break, since the leak detection system itself is non-safety-related, no credit can be taken for the successful performance of this function.

The team considers the following issues to be unresolved items subject to further review by the Staff:

- The use of the leak-before-break concept as a basis for not providing environmental qualification for safety-related equipment.

- The reliance of a safety-related system upon non-safety-related equipment in order to perform safety functions.

BASIS: 10 CFR 50.49 requires safety-related equipment to be qualified for environments imposed during and following design basis events.

REFERENCES

1. SNED Modification Approval Letter, "M12-2(3)-84-17 Installation of the Reactor Water Clean-up Room Leak Detection System," June 13, 1984.
2. NUREG-0823, Integrated Plant Safety Assessment, Systematic Evaluation Program, Dresden Nuclear Power Station, Unit 2, Commonwealth Edison Company, Docket No. 50-237, Final Report, February, 1983.
3. Nutech Engineers, Inc. Report, "Leak Before Break Justification, Reactor Water Clean-up System, Dresden Units 2 & 3, Quad Cities Station Units 1 & 2, May 24, 1984."

D2.1-4 (Deficiency) Reactor Water Clean-up Leak Detection System Parameters

DESCRIPTION: A Nutech report (Reference 2) includes calculations to determine critical crack size for through wall leakage without plastic or catastrophic failure of the piping considered. Table 1 of the report lists critical crack sizes and calculated flow rates for several pipe sizes. For a 10" pipe diameter, a flow rate of 58.9 gpm is anticipated through the critical crack determined. The report concludes that the proposed leak detection system is adequate since the calculated leak rate is, "much larger than necessary to trigger the proposed room temperature monitors." However, this conclusion is not substantiated in the report since the leak rate to be detected by the system instrumentation is not referenced in the report.

In a letter to SNED (Reference 3), Nutech described the rationale for the 150°F trip set point for the reactor water clean-up leak detection system, indicating that a leak of 5 gpm will result in a 30°F rise in room temperature and references a Bechtel calculation (Reference 4) to support the selected leakage criteria. The results of this calculation are based on a leak rate of 9775 lb/hr (26.3 gpm) which is not consistent with the 5 gpm flow described in the Nutech letter (Reference 3) but is conservatively less than the anticipated flow (58.9 gpm) resulting from the Reference 2 critical crack size/flow calculation for the 10" diameter RWCU line involved.

The team found no documented basis for the 5 gpm flow rate which Nutech had established as the design basis flow for the leak detection system. Subsequent to the inspection, Bechtel indicated to the team that the 5 gpm is based on General Electric BWR Generic Technical Specifications requiring leakage detection inside containment for moderate energy through wall leakage cracks defined in BTP MEB 3-1 (Reference 6). Bechtel stated that the 5 gpm leakage corresponds to a .01 square inch leak which is a factor of 100 less than the flow area calculated in accordance with BTP MEB 3-1 for the 8 inch schedule 80S pipe Dresden line (Reference 7 indicates the correct RWCU line size is 10 inches).

Subsequent to the inspection, Bechtel adjusted the Reference 4 calculation to account for the 5 gpm flow rate. The resultant temperature rise above ambient for 5 gpm is significantly less than that originally calculated for 26.3 gpm. Additional information provided by Bechtel indicated that the gross free volumes for several areas affected by the temperature transient were compared to determine the most conservative volume to be used in the transient. However, no documented evidence of these calculations were provided. Bechtel did not address venting of one area to another via any interconnections between these areas. Bechtel referred to the results of the Reference 4 calculations as "estimated" or "order-of-magnitude values" upon which the design of the system was based.

Reference 3 indicated that the 150°F leak detection setting is based on a maximum ambient room temperature of 120°F plus the 30°F temperature rise calculated in Reference 4. However, it was not apparent that this setting gave consideration to ambient room temperatures covering all system modes of operation. In its review of Reference 4, the team identified several deficiencies related to this and other aspects of the leak detection system design basis:

1. The initial conditions for the transient calculation include pressure, temperature, and relative humidity and are established in the calculation. However, there is no documented basis given for the stated parameters other than reference to conditions "agreed upon" subsequent to discussions with the project civil group. The calculation states that the 90°F temperature used in the calculation as the initial condition, "does not represent the maximum possible room temperature during normal plant operation, but is more representative of the normal room temperature." It is not clear whether this 90°F temperature is representative of all modes of operation. If so, the above 150°F setting, which is based on 120°F ambient temperature, may be unconservative.

2. It appears that this calculation is a civil/structural calculation to determine the effects of room pressurization following a small break in the reactor water clean-up piping. A number of assumptions, intended to maximize pressure and temperature to assure conservative assessment of block wall design, are not conservative relative to temperatures to be sensed by the leak detection system. For example:

   a. The calculation assumes, "no venting of the room to the rest of the building," and does not account for the effects of HVAC in the area. Reference 5 indicates that temperature rise is "relatively insensitive to HVAC performance," but there is no documented basis for this conclusion.

   b. A 10% reduction in room free volume is assumed; no basis for the 10% reduction was given. Reduction in room free volume will produce higher temperature results.

   c. Heat sinks (concrete walls, ceiling, floor) were modeled to limit heat transfer, thereby increasing temperature.

3. No consideration was given to drainage (through doorways, etc.) from the areas where leakage may collect and its effect on reduced temperatures during the transient.

4. The "Results" of the calculation indicate that the room temperature transient will be affected by vent paths which allow the room to depressurize. However, no quantitative evaluation of the effect of such paths is documented.

The above issues concerning inconsistencies in room temperatures and leak rates and unconservative temperature transients reflect upon inadequate interfaces between SNED, Bechtel, and Nutech, and Nutech's inadequate review of design input from Bechtel.

BASIS: The licensee has committed to Regulatory Guide 1.64 (References 9 and 10) which endorses ANSI N45.2.11 (Reference 8). This requires the design to be traceable to the original design input. Contrary to this requirement, there was an inadequate documented basis for the temperature trip set point for the reactor water clean-up leak detection system.

REFERENCES


1. SNED Modification Approval Letter, "M12-2(3)-84-17 Installation of the Reactor Water Clean-up Room Leak Detection System," June 13, 1984.
2. Nutech Engineers, Inc. Report, "Leak Before Break Justification, Reactor Water Clean-up System, Dresden Units 2 & 3, Quad Cities Station Units 1 & 2, May 24, 1984."
3. Nutech Engineers, Inc. Letter to SNED, "Dresden Station Units 2 & 3 RWCU Leak Detection System, COM-78-061, May 2, 1985."
4. Bechtel Calculation No. C9K-0015, "Transient Response of Reactor Water Clean-up System (RWCUS) Heat Exchanger Room to a Small Pipe Break," Revision 2, June 8, 1984.
5. Bechtel Letter to SNED, "Dresden Units 2 and 3 and Quad Cities Units 1 and 2, Temperature and Radiation Monitors," G35-11-084, November 18, 1982.
6. U. S. Nuclear Regulatory Commission Standard Review Plan 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment," Revision 1, July, 1981.
7. Sargent & Lundy Drawing M-361, Dresden Nuclear Power Station Unit 3 Diagram of Reactor Water Clean-up Piping, Sheet 1, Revision AE, September 7, 1985.
8. ANSI N45.2.11, Quality Assurance Requirements for the Design of Nuclear Power Plants, 1974.
9. Commonwealth Edison Company Quality Assurance Manual, Quality Requirement 2.0.
10. Regulatory Guide 1.64, Quality Assurance Requirements for the Design of Nuclear Power Plants, 1974.

O2.2-1 (Observation) HPCI Room Cooler Modification

DESCRIPTION: The team reviewed several modifications from previous outages which might be related to work done during the current outage, including a change to the high pressure coolant injection (HPCI) room cooler piping to permit flow reversal through the cooler. This modification (M12-3-81-31) was similar to a modification to the containment cooling service water pump vault coolers (M12-3-84-14) being developed for the current outage, but not yet complete. The change consisted of the addition of two valves and a short portion of piping to the existing HPCI room cooler. This 2" piping is designated Class 1 piping as defined in Section 12 of the FSAR.

In a letter to SNED (Reference 1), Sargent & Lundy indicated that the HPCI room cooler piping was originally analyzed (and support loads were determined) using simplified piping curves, and the addition of these valves close to the room cooler had no significant impact on the original analysis of this line. However, the team could find no documented evidence as to whether and how much the addition of these valves would increase loads on the piping supports. Further, the Sargent & Lundy letter indicated that the field had been advised, "to increase the rod size of the existing support at this location from 1/2 inch to 3/4 inch size to accommodate any increase in load due to the additional valves." The increased rod size had been installed in the plant. In the absence of an analysis, it is not clear whether the size increase was adequate. The licensee's engineering decision should be supported by analysis.

REFERENCES

1. Sargent & Lundy Engineers Letter to SNED, "HPCI Room Cooler Modification," March 23, 1982.
2. Commonwealth Edison Company Quality Assurance Manual, Quality Requirement 2.0.
3. Regulatory Guide 1.64, Quality Assurance Requirements for the Design of Nuclear Power Plants, 1974.
4. ANSI N45.2.11, Quality Assurance Requirements for the Design of Nuclear Power Plants, 1974.

D3.1-1 (Deficiency) Reactor Building Service Penetrations

DESCRIPTION: The subject modification (References 1,2) provides for the installation of twelve five inch diameter penetrations in the Dresden Unit 3 reactor building exterior wall. These penetrations are intended to allow services such as gas, electric and welding leads to be run into the reactor building during the Dresden Unit 3 replacement of stainless steel recirculation pipe subject to intergranular stress corrosion cracking (Reference 3).

The team reviewed the Impell calculation which qualifies the reactor building penetrations (Reference 4), and the Impell drawings which detail the design and fabrication of the pipe penetrations (References 5,6), and noted the following:

(1) The calculation does not consider the effects of pressure due to a high energy line break (HELB), a conservative estimate of which can be obtained from the environmental zone maps prepared by Bechtel for CECo in response to NRC IE Bulletin 79-01B (Reference 7). Figure 2 of the referenced specification tabulates a HELB pressure of 21 psia at the location of the reactor building penetrations. The calculation does consider the reactor building operating pressure differential of 0.25 psi which is specified in FSAR Section 5.3.1. However, the magnitude of the pressure differential due to a HELB is approximately twenty times greater.

(2) The piping penetrations will be capped with blind flanges when no longer required. A lead shield plug will be placed in each penetration to approximate the radiation shielding of the 5 inch diameter concrete cores which have been removed from the reactor building wall. However, no restraint is provided for the lead shield plug (Reference 5, Detail C), and the calculation does not consider the potential effects of movement within the piping penetration due to a seismic event.

(3) Drawing SK-5007 (Reference 5) does not specify the material required for the 150 lb. blind flanges to be used to cap the pipe penetrations.

(4) Drawing SK-5007 does not specify painting or galvanizing for the flanges, studs and nuts used to cap the exterior segments of the pipe penetrations.

(5) Drawing SK-5008 (Reference 6, Detail A) does not specify a minimum edge distance from the pipe penetration shear lugs to the face of the reactor building wall. Credit for the axial restraint provided by the shear lugs (page 6 of the calculation) cannot be taken without specifying a minimum edge distance.

BASIS: The licensee has committed to Regulatory Guide 1.64, which endorses ANSI N45.2.11. This requires consideration of applicable design inputs such as design pressures. Contrary to this, the calculation did not consider pressure due to HELB.

With respect to items 2-5, the CECo Standard Quality Articles appended to each bid specification require that the contractor quality assurance program meet the requirements of 10CFR50, Appendix B, Criterion III, Design Control, which requires that measures be established to assure that the design basis be correctly translated into specifications, drawings, procedures and instructions.

REFERENCES


1. CECo Modification Approval Letter for Modification M12-3-85-53, Penetrations in Reactor Building, AIR 12-85-07, dated July 2, 1985.
2. CECo Addendum No. 1 to Modification Approval Letter for Modification M12-3-85-53, Penetrations in Reactor Building, AIR 12-85-07, dated August 12, 1985.
3. CECo Modification Approval Letter for Modification M12-3-85-16, Reactor Recirculation System Pipe Removal and Replacement, AIR 12-85-07, dated September 10, 1985.
4. Impell Calculation No. 0595-218-1, IGSCC Support Penetration Details, Job No. 0595-218-1451, Rev. 3, dated November 7, 1985.
5. Impell Drawing No. SK-5007, IGSCC Support and Torus Recoat Penetration Details/Dresden Station/Commonwealth Edison Company, Sheet 1 of 1, Rev. 2, dated November 13, 1985.
6. Impell Drawing No. SK-5008, Auxiliary Power Penetration Details/ Dresden Station/Commonwealth Edison Company, Sheet 1 of 1, Rev. 2, dated November 13, 1985.
7. Bechtel Specification N102, Job No. 13524-068, Response to IE Bulletin 79-01B/Procedure for Use of Environmental Zone Maps for Dresden Nuclear Power Station Units 2 and 3/Commonwealth Edison Company, Rev. 0, dated April 10, 1985.

03.1-2 (Observation) 125 Volt Battery Rack Seismic Qualification

DESCRIPTION: CECo Modification Approval Sheet No. M12-3-85-25 authorized the replacement of the Dresden Unit 3 125 volt battery with a battery of larger capacity and the replacement of the existing battery rack with a seismically qualified battery rack. However, the Sargent & Lundy design for the new Dresden Unit 3 battery rack was not complete at the time of the inspection. The team therefore reviewed the Sargent & Lundy design for the Dresden Unit 2 125 volt battery rack (CECo Modification No. M12-2-85-25).

The team reviewed the Sargent & Lundy battery rack structural analysis (Reference 6) and the Sargent & Lundy battery rack design drawing (Reference 7). The span lengths for the three Unistrut P1000 base stringers which support the batteries were 2 ft.-5 1/4 in. Page 6 of the referenced Sargent & Lundy calculation specifies a design criterion of 33 Hz fundamental frequency for the battery rack. Free vibration analysis of the battery rack analytical model yields a fundamental frequency of 34.5 Hz, therefore, satisfying the 33 Hz minimum frequency criterion. However, the fundamental frequency of the Unistrut P1000 base stringers was not calculated because the battery masses were lumped at the joints of the battery rack analytical model. In conversation with Sargent & Lundy, the team noted that the battery rack could not be considered rigid without a formal check of the base stringer fundamental frequency. To address this issue, Sargent & Lundy prepared an addendum to the original calculation (Reference 8). The base stringer fundamental frequency calculated for this revised analysis is 26.3 Hz, which is less than the minimum frequency criterion of 33 Hz. However, an examination of the design response spectra contained in Sargent & Lundy Specification T-3349 (Reference 9) indicates that ZPA acceleration values govern for frequencies greater than approximately 25 Hz, so that the battery rack can be considered seismically rigid. The team notes that the addition of a fourth base stringer would raise the base stringer fundamental frequency to approximately 30 Hz.

REFERENCES

1. Sargent & Lundy Letter to CECo, Project No. 0300-14, dated May 20, 1985.
2. Gould Drawing No. 078437C, Layout for 3 Cells NAX/NCX-1680 on Special Test Rack S07-078437-876, Sheet 7 of 7, Rev. A, dated October 7, 1981.
3. Structural Design of Test Rack for 3 Cells of NAX/NCX-1680 Gould Batteries, Job No. 881, Prepared by Ralph C. Dumack, P.E. & Associates, no date.
4. Sargent & Lundy letter to CECo, Project Nos. 7328-00 through 7331-00, 7329-30, 7330-30, dated October 24, 1985.
5. Structural Calculation Prepared by ARORA and Associates, P.C. for Gould, Job No. 1034, Battery Rack, dated April 29, 1985.
6. Sargent & Lundy Calculation No. 7294-TB-05, 125V Battery Replacement Trolley Beam and Rack Design, Project No. 7294-00, Unit No. 2, Rev. 0, dated September 26, 1985.
7. Sargent & Lundy Drawing No. B-1892, 125V DC Battery Rack/Plans Elevations, Sections and Details/Dresden Nuclear Power Station-Unit 2/Commonwealth Edison Company, Rev. A, dated September 24, 1985.

8. Sargent & Lundy Calculation No. 7294-TB-05, Battery Rack Loads, Project No. 7294-00, pages 24.1-24.3, dated January 9, 1986.
9. Sargent & Lundy Specification No. T-3349, Certification of Specification for Storage Batteries and Accessories/Dresden Station-Units 2 and 3 and Quad Cities Station-Unit 1/Commonwealth Edison Company, Rev. Add. 1, dated August 7, 1985.

D3.1-3 (Deficiency) Instrument Rack Modifications

DESCRIPTION: The subject modification (References 1,2) replaces existing instruments with instruments qualified to the environmental criteria in NRC IE Bulletin 79-01B and IEEE 323-1974, and the seismic criteria in IEEE 344-1975 (Reference 3). The team reviewed the in-situ seismic qualification of the instrument racks on which new seismically qualified instruments have been mounted, and the generation of required response spectra (RRS) at these instrument mounting locations. The new instruments are considered seismically qualified for installation at Dresden Units 2 and 3 if the RRS computed at the instrument rack mounting locations are enveloped by the instrument test response spectra (TRS). CECo authorized Cygna Energy Services to seismically qualify thirty instrument racks currently installed in Dresden Units 2 and 3 and Quad Cities Units 1 and 2 (Reference 4).

Cygna performed a walkdown to define the instrument rack geometry and instrument locations, performed some in-situ vibratory testing of the racks, created analytical models of the instrument racks, generated response spectra at the mounting locations of the new instruments, and issued drawings which detailed modifications to the instrument racks. The team noted the following:

(1) The walkdown packages for instrument racks D2203-19A and -19B were not prepared in accordance with the governing Cygna work instruction (Reference 5). Rack structural dimensions were not completely documented; existing weld sizes were not documented; fixity of adjacent rack bays was not documented; and instrument weights were not tabulated (Reference 6). As a consequence, the accuracy of the rack analytical models incorporating this data could not be readily checked.

(2) Cygna performed in-situ testing of instrument racks 2202-6 and -8 installed at Quad Cities Unit 2 to compute rack modal frequencies, modal damping ratios and mode shapes to be "used to facilitate the synthesis and verify the accuracy of the finite element models used for the seismic qualification of the instrument racks" (Reference 7). The results of the in-situ testing for racks 2202-6 and -8 proved to be inconclusive (Reference 8). The results of the in-situ testing of rack 2202-6 are summarized on page 2 of the referenced calculation: "Due to the complex nature of this rack, it was difficult to obtain well defined modes. Internal parts of the instruments tended to rattle about, attached conduits and piping were not well supported, and the attached 1/8 in. mounting plates were not tight to the frame work, and tended to form opening and closing gaps in the system, causing a distortion of the linear response expected by the impulse-response test method. The non-linear response resulted in the frequency peaks not being well defined, and difficulty was encountered in obtaining reasonable estimates of damping. Measured values ranged from 3.9 % to less than 0.5 % of critical." In-situ testing of rack 2202-8 was similarly inconclusive. Section 2.2 of NUREG-1030 (Reference 9), Development and Assessment of

In-Situ Testing Methods to Assist in Qualification of Equipment, should be reviewed by CECo with respect to the above matters. It appears that the 1/8 in. thick instrument mounting plates for instrument racks 2202-6 and -8 are intermittently rather than continuously welded to the rack frame. The mounting plate therefore generates local "noise" under excitation that is not accounted for in the instrument rack analytical model, which assumes that the mounting plate is rigidly connected to the instrument rack.

(3) Page 11 of the Reference 8 calculation set notes that rack 2252-30B is adjacent to, but not physically attached to the side channel of bay A of rack 2202-8. The walkdown package for rack 2202-8 documented a gap of 1/8 in. between the racks. However, the maximum in-plane displacement at the top of rack 2252-30B due to seismic load was not evaluated with respect to the 1/8 in. gap separating the racks to confirm that no seismic interference between the racks will occur. The seismic analysis performed for rack 2202-8, which assumes that the rack is free-standing and not subject to seismic interference from adjacent equipment, is therefore inconsistent with as-built conditions.

(4) Instrument rack as-built drawings were not prepared in accordance with the governing Cygna work instruction (Reference 10), which requires that the major structural components be dimensioned, and that instrument weights be tabulated. The as-built drawings for racks 2203-19A and -19B did not specify the rack plan dimensions, or tabulate the instrument weights (References 11,12).

(5) Several of the instrument racks in the Cygna scope of work appear to be subject to relatively high temperatures due to a high energy line break (HELB). New equipment for instrument racks 2203-7 and -8, for example, is qualified to a HELB temperature of 262°F. However, the instrument racks were not evaluated for this thermal load (Reference 13).

(6) Because the Cygna in-house computer program used to perform the instrument rack seismic analysis cannot model two-dimensional (plate) elements, the instrument rack mounting plates are approximately modeled as a grid of beams. The team requested the specific examples used to validate this approximation. One example had been performed on the in-house program, and compared with an ANSYS run. The results were within eight percent. However, neither example considered asymmetric loading locations and magnitudes, out-of-plane loads, or provided solutions for vertical and in-plane seismic excitation. Inaccurate modeling of the mounting plates would invalidate the response spectra generated at locations on the mounting plates.

(7) Cygna is not documenting the seismic qualification of the new instruments with respect to the seismic response spectra computed at instrument rack mounting plate locations.

(8) The keyplan for rack D2203-19A (Reference 14) incorrectly locates the Dresden Unit 2 instead of the Dresden Unit 3 rack.

(9) End returns on fillet welds are not being detailed on the Cygna design drawings as required by the AISC Code (Reference 15).

BASIS: Items (1), (4) and (9) indicate non-compliances with internal procedures and the AISC Code. Items (3), (5) and (6) indicate failure to address pertinent design inputs, as required by ANSI N45.2.11.

REFERENCES


1. CECo Modification Approval Letter for Modification M12-2(3)-83-40, Replacement of Yarway Reactor Water Level Switches, AIR 12-82-67, dated June 6, 1984.
2. CECo Addendum to Modification Approval Letter for Modification M12-2(3)-83-40, Combining Functions of Transmitters 2(3)-1501-55 A&B with 2(3)-1543 A&B - Dresden Station Units 2 and 3, dated February 25, 1985.
3. CECo Modification Approval Letter for Modification M12-2(3)-83-40, Replacement of Yarway Reactor Water Level Switches, and Modification M12-2(3)-83-36 through -39, Replacement of Pressure, Differential Pressure, and Floor Switches, Dresden Station Units 2 and 3, AIR 12-82-67, dated June 6, 1984.
4. CECo Bid Specification IR-1, Engineering Services to Develop and Analyze the Seismic Response of Instrument Racks at Dresden Units 2 & 3 and Quad Cities Units 1 & 2, Rev. 0, no date.
5. Cygna Work Instruction for Collection of As-Built Data, WI-1, Rev. 1, dated October 21, 1983.
6. Cygna Walkdown Package for Instrument Racks D2203-19A and -19B, Dresden Unit 3, Job No. 83078, dated October 18, 1983.
7. Cygna Work Instruction for Low Amplitude Vibration Testing of Instrument Racks, WI-2, Rev. 0, dated October 14, 1983.
8. Cygna Calculation Set No. 001, Results of In-Situ Testing at Quad Cities Unit 2, Instrument Racks 2202-6 and 2202-8, Job No. 83078, File No. 331F, Rev. 0, dated December 3, 1983.
9. NUREG-1030, Seismic Qualification of Equipment in Operating Nuclear Power Plants/Unresolved Safety Issue A-46/Draft Report for Comment, published August, 1985.
10. Cygna Work Instruction for Analysis of Instrument Racks, WI-3, Rev. 0, dated November 15, 1983.
11. Cygna Drawing No. SK-D2203-19A, Instrument Rack No. 2203-19A, Rev. 0, dated December 5, 1985.
12. Cygna Drawing No. SK-D2203-19B, Instrument Rack No. 2203-19B, Rev. 0, dated December 5, 1985.
13. Bechtel Specification N102, Job No. 13524-068, Response to IE Bulletin 79-01B/Procedure for Use of Environmental Zone Maps for Dresden Nuclear Power Station Units 2 and 3/Commonwealth Edison Company, Rev. 0, dated April 10, 1985.
14. Cygna Drawing No. M-3628, Modification Designs for Instrument Racks D2203-19A & D2203-19B, Sheet 1 of 3, Rev. 0, dated December 21, 1984.
15. AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, Paragraph 1.17.7, Eighth Edition.

03.1-4 (Observation) Failure to Reference Original Design Specifications

DESCRIPTION: Some modification packages did not reference the Sargent & Lundy and General Electric design specifications which governed the original design and procurement of NSSS and BOP piping systems and equipment. For example:

1. The Impell calculation which qualifies the temporary service reactor building penetrations for Modification M12-3-85-53 does not reference an original piping penetration design specification, although GE apparently prepared such a specification. During a site visit on January 8, 1986, the team accessed General Electric Specification 22A1167, Design Specification for Penetrations-Primary Containment, Rev. 0, dated May 8, 1967. Section 4.2 of the specification refers to another General Electric design specification related to the primary and secondary containment.

2. The Cygna instrument rack calculations for Modification M12-23(3)-83-40 do not reference an original Sargent & Lundy design specification, although the team confirmed the existence of Sargent & Lundy Design Specification K-2264, Instrument Racks, during a site visit.

3. The specification for the expansion bellows, which Impell prepared as part of IGSCC replacement piping Modification M12-3-85-16, does not reference the previously cited General Electric design specification for primary containment penetrations, although Section 4.4.3, Bellows Expansion Joints, provides detailed expansion bellows design requirements and configuration types.

In order to ensure that design work associated with modification packages is consistent with the original design basis, it is important to review the design specifications for the original design when evaluating the modification package. To ensure that this process occurs, it is prudent for the modification package to reference the original design specifications. The team recommends that CECo procedures be revised to make this a requirement.

D3.1-5 (Deficiency) IGSCC Pipe Removal and Replacement

DESCRIPTION: The subject modification (Reference 1) authorizes the replacement of critical piping susceptible to intergranular stress corrosion cracking (IGSCC) at Dresden Unit 3. The team reviewed the Impell specifications and procedures prepared for this modification, and reviewed the modification of LPCI Loop B pipe supports M-1200D-1003, and -1004 (References 3,4). The team noted the following:

(1) Impell design specification 3 (Reference 5), references the Sargent & Lundy piping specification originally prepared for Dresden Units 2 and 3 (Reference 6). Section 24, item E of the standard piping specification appended to the Sargent & Lundy piping design specification requires in part that all straps and clamps attached to stainless steel pipe be of the same material as that specified for the pipe. However, the replacement supporting plate and U-bolt for support M-1200D-1004 are fabricated from carbon steel. This appears to be Impell standard practice for this modification.

(2) The pipe support configuration for spring hanger M-1200D-1004 was modified upon replacement; the original and modified designs are shown on the Reference 7 and 8 drawings. The pipe saddle originally supporting the 16 inch LPCI Loop B pipe was replaced by a U-bolt and a 6 x 22 x 1 inch plate, which is supported by a 6 inch segment of 4 x 4 x 1/4 inch tube steel. An adjacent snubber originally suppressed the horizontal seismic motion, but has been deleted for the new design. The replacement support design appears to be standard Impell practice for this modification. Impell did not compute the stresses in the plate due to the pipe support inertial forces caused by combined horizontal and vertical design basis earthquake. This calculation is necessary because the replacement plate does not have the same load capacity as the original pipe saddle. The calculated seismic displacements for the pipe at the hanger support location are approximately 1/8 inch (+/-) vertically and 3/8 inch (+/-) laterally.

BASIS: Impell did not specify replacement stainless steel straps and clamps as required by the Sargent & Lundy piping design specification. Contrary to ANSI N45.2.11 requirements for considering design input such as seismic loads, Impell's replacement pipe support saddle was not evaluated for seismic loads.

REFERENCES

1. CECo Modification Approval Letter for Modification M12-3-85-16, Reactor Recirculation System Pipe Removal and Replacement, AIR 12-85-07, dated September 10, 1985.
2. Impell Specification No. 0595-010-S-01, Specification for Removal and Replacement of IGSCC Susceptible Piping/Commonwealth Edison Company, Rev. 1, dated March 8, 1985.
3. Impell Drawing No. D3-RRCI-RP04, Dresden 3 Piping Replacement Demolition Layout Drawing Recirculation Loop B (RRCI, LPCI, SDC), Sheet 2 of 3, Rev. 1, dated October 18, 1985.

4. Impell Drawing No. D3-RRCI-RP02, Dresden 3 Piping Replacement/Mod. No. M12-3-85-16, Sheet 2 of 3, Rev. 1, dated October 1, 1985.
5. Impell Specification No. 0595-010-S-03, Design Specification for the Replacement of IGSCC Susceptible Piping Project-Dresden Station Unit 3, Rev. 2, dated September 13, 1985.
6. Sargent & Lundy Specification K-2202, Piping Systems Dresden Units 2 and 3, July 12, 1967 through Supplement 6.
7. Chicago Bridge & Iron Company (CBI) Drawing No. 552, Removal/ Installation Interference #552/Existing Pipe Support B3-1502/New Pipe Support M-1200D-1004, Sheet 1 of 2, Rev. 1, dated October 7, 1985.
8. Impell Drawing No. M-1200D-1004, Hanger Mark No. M-1200D-1004, Rev. 0, dated May 30, 1985.

03.1-6 (Observation) Battery Charger Specifications

DESCRIPTION: The original Sargent & Lundy specification for the battery chargers (Reference 1) did not specify seismic criteria for the battery chargers, and it appears that the original battery chargers procured for Dresden Units 2 and 3 were not purchased as safety-related equipment. Battery chargers 3 (250 V dc) and 3A (125 V dc) replaced the original chargers around 1979-80. The replacement 125V dc battery chargers for Dresden Units 2 and 3 were purchased as safety-related equipment (Reference 2). A copy of the Sargent & Lundy mechanical and electrical equipment list dated January 18, 1980 classifies battery chargers 3 and 3A as safety-related (Reference 3).

The Sargent & Lundy mechanical and electrical equipment list references the Sargent & Lundy design specifications used to procure the original battery chargers for Dresden Unit 3, rather than revised specifications subsequently used to purchase replacement battery chargers. The 125 V dc battery chargers for Dresden Units 2 and 3 were actually purchased to a CECo design specification (Reference 5). The team recommends that the list be corrected to reflect current installed equipment.

REFERENCES

1. Sargent & Lundy Specification K-2212, R-2352, Storage Batteries and Chargers, dated January 31, 1967, revised April 19, 1967.
2. CECo Purchase Order No. 223975, Dresden-Units 2 & 3/Install Redundant 125 Volt Charger, dated May 10, 1978.
3. Sargent & Lundy Mechanical & Electrical Equipment List, Commonwealth Edison Company Dresden Station Unit 3, 5995-00, page 26, dated January 18, 1980.
4. Sargent & Lundy Letter to CECo, Project No. 6129-00, dated May 9, 1984, with attached Electrical Equipment Anchorage Study.
5. CECo System Standard No. EM-12925, Battery Chargers, dated January 1, 1971.

D4.1-1 (Deficiency) Separation of Redundant Class 1E Channels and Separation of Wiring for Safety-Related and Non-Safety-Related Circuits for Analog Transmitter Replacement Modifications

DESCRIPTION: The original BWR design used direct acting differential pressure switches for automatic reactor trip and containment isolation of main steam lines, based on measurements of reactor vessel low water level. Dresden 3 and other BWR plants have replaced these level switch sensors with electronic transmitters.

Four reactor vessel water level sensor channels were used for these functions, and were assigned to four distinct separation divisions: A1, A2, B1, and B2 (Reference 1). Since each channel de-energizes for its trip function, redundant channels A1 and A2 must be kept separated from one another and redundant channels B1 and B2 must also be kept separated from one another. In the original design, two three-bay panels having internal barriers were used to maintain the required channel separation. To reduce the probability of spurious reactor trip, one power source was used for the "A" channels and another power source for the "B" channels.

Nutech developed a major portion of the design modification to convert the reactor vessel water level switches to analog transmitters (Reference 2). Based on the team's discussions with Nutech personnel, the initial design modification had channels A1 and B1 located in one cabinet and connected to one power source. Channels A2 and B2 were located in another cabinet and were connected to another power source. This design met the separation criteria, but because it did not meet the power source assignment scheme of the original reactor protection system design, the probability of spurious reactor trip was significantly increased. As a result of a Nutech meeting with Quad-Cities station personnel on June 11, 1984, this design was subsequently modified to locate channels A1 and A2 in one cabinet and B1 and B2 in the other cabinet (References 3, 4, and 5). This same design was reviewed by Nutech and Dresden Station personnel on June 25, 1984. This change corrected the spurious trip issue; however, it introduced a violation of channel separation criteria within each cabinet.

In the present design configuration a non-mechanistically postulated single event, for example, a fire as considered in the original design (Reference 1), could disable both redundant reactor vessel water level trip outputs in either cabinet. Since there is no diversity in the form of a backup variable to reactor vessel water level provided in the design, this postulated single failure is a violation of the single failure criterion stated in IEEE Std. 279-1968, which is listed as an applicable standard for the design modification (Reference 6).

The modification approval letter (Reference 6) identified that the design modification was intended to be consistent with the design provided by General Electric Licensing Topical Report NEDO-21617A. However, the design modification differed from this Topical Report and the original design approach in its use of two rather than four cabinets as well as in the coordination of process sensor physical connections with the safety system

coincidence logic. Such coordination is needed to assure compliance with single failure and separation criteria. In the transmitter modification, reactor water level channels A1 and A2 have been located together in the 2203-73B division II panel (Reference 4) and the B1 and B2 channels have been located together in the 2203-73A division I panel (Reference 5). The Nutech design modification violates separation criteria (Reference 1) in that redundant channels A1 from rack 2203-5 and A2 from rack 2203-6, as well as B1 from rack 2203-5 and B2 from rack 2203-6, have not been kept separate and independent of one another.

To assure proper separation of safety and non-safety wiring, note 2 on panel wiring drawings required the use of an asterisk (*) to denote which non-safety-related wiring required separation. In addition, one modification combined the independent functions of one safety-related switch and one non-safety-related switch into a single transmitter instrument loop. From the team's review of the panel wiring diagrams with Nutech personnel, it was determined that, for at least two modifications (References 7 and 8), an asterisk had not been used to denote non-safety wiring requiring separation (References 4 and 5). Drawings prepared for the combined transmitter channel also did not indicate the required separation of annunciator wiring from safety-related wiring in the trip portion of the channel (Reference 9).

BASIS: For the transmitter design modification involving four reactor vessel water level channels, the required channel separation criteria (Reference 1) were violated by locating redundant channels that must be kept independent of one another in close proximity in each of the two cabinets. This design configuration also violated the separation requirements described in Dresden 3 FSAR Amendments as well as those stated in IEEE Proposed Standard 279-1968, and paragraph 5.1.2 of the General Electric Licensing Topical Report NEDO-21617A (Reference 6).

Nutech drawings did not depict the separation of non-safety-related wiring relative to safety-related wiring in accordance with an applicable drawing note, and did not identify needed separation provisions for the combining of independent safety and non-safety switches into one transmitter loop.

REFERENCES

1. General Electric Specification, 22A2501, "Separation Requirements for Reactor Safety and Engineered Safeguards Systems," Rev. 0, 1/28/69.
2. Dresden 3 Design Modification M12-3-83-40, Reactor Vessel Water Level Switches to Level Transmitters LT-3-263-57A and B, LT-3-263-58A and B, and LT-3-263-73A and B.
3. Nutech Loop Schematic/Functional Block Diagram - Analog Trip System Class 1E Instrumentation Upgrade, 12E-7820, Rev. E, 8/22/85.
4. Nutech Wiring Diagram 12E-7861, Rev. D, 8/19/85.
5. Nutech Wiring Diagram 12E-7860, Rev. D, 10/7/85.
6. CECo Modification Approval Letter, M12-3-83-36 through -40, 6/6/84.
7. Dresden 3 Design Modification M12-3-83-37, Pressure Suppression Differential Pressure Switches to Differential Pressure Transmitters DPT-3-1622A and B and Level Switch to Level Transmitter LT-3-1626.
8. Dresden 3 Design Modification M12-3-83-39, RHR Flow and Differential Pressure Switches to Flow Transmitters FT-3-1501-58A and B and Differential Pressure Transmitters DPT-3-1543A and B.
9. Sargent and Lundy schematic diagram 12E-3495, LPCI Containment Cooling System, Rev. P, 8/19/85.

D4.1-2 (Deficiency) Reliability, Failure Modes, and Response Time Requirements for Analog Transmitter Replacement Modifications

DESCRIPTION: The original BWR design used direct acting pressure and differential pressure switches for automatic initiation of reactor protection, containment isolation, and emergency core cooling systems. Dresden 3 and other BWR plants have replaced some pressure and level switch sensors with electronic transmitters to obtain improved performance and to reduce the frequency of sensor calibration.

Substitution of a transmitter for a given switch is not a "one-for-one" replacement since a number of additional components are required for transmitter channels as compared to direct acting pressure and level switches. As a result of the substitution, evaluations need to be performed to account for differences in sensor channel component estimated reliability (as described in the USAR), postulated failure modes for added components, and sensor channel response time. The team reviewed each of the 21 transmitter substitutions specified for Dresden 3 (References 2 through 6), and identified that there was no documented evidence that reliability, failure mode and effects analysis, and response time analyses were performed. The Commonwealth Edison modification approval letter (Reference 9) stated that the intended design "changes are consistent with General Electric Licensing Topical Report NEDO-21617A"; however, the team was unable to confirm that equipment supplied for the Dresden 3 plant by Nutherm was identical in all respects to the requirements specified by General Electric purchase part drawings listed in the Topical Report. Hence, the Dresden 3 applicability of various General Electric analyses contained in this report was not confirmed.

Nutech engineers developed a major portion of the five design modifications to convert certain reactor protection system and emergency core cooling system initiation sensor switches to analog transmitters. The team noted the following deficiencies regarding reliability, failure mode and effects analysis, and response time assessment for this transmitter design modification:

(1) Nutech indicated that quantitative reliability values for the transmitters were not specified even though minimum reliability values for the original switches were provided in Dresden 3 FSAR Figure 6.2.37. This availability analysis for a small line break included a 0.99985 reliability estimate for four reactor water level sensor channels and a 0.99968 reliability estimate for four drywell pressure sensor channels. Based on the coincidence logic reliability values, a single channel was estimated to have a minimum 0.99996 and 0.99992 reliability value respectively. A confirmatory analysis was not performed to assure that equivalent or better reliability values were provided by the replacement equipment (in comparison to the Dresden 3 FSAR reliability values), as required by IEEE 352-1975, which was

listed as an applicable standard in the Commonwealth Edison modification approval letter (Reference 8).

(2) Based on Nutech responses, transmitter failure mode and effects analyses were performed but were not documented. The team reviewed several draft FMEA's prepared by Nutech during the inspection, and determined that they lacked system level impact evaluations specified by IEEE Std. 352-1975. There was no evidence that additional internal failure modes introduced by the design modification had been considered for the following equipment:

(a) scaling amplifier in the master trip unit;

(b) internal power supplies in the slave trip unit for nine channels;

(c) internal fusing and diode in the 24 volt dc power supply in the slave trip unit; and

(d) trip adjustment potentiometer, trip comparator, and trip output circuit in the slave trip unit.

Each of the identified components could cause the slave relay output to differ from the master relay output (Reference 7).

(3) The effect of sensor response time differences on the FSAR accident analyses was not assessed, contrary to a design basis requirement stated in Section 3 of IEEE Std. 279-1968 (Reference 9).

BASIS: For the transmitter design modification, no reliability requirements were specified for the transmitter, power sources, master trip unit, slave trip unit, and their output relays. There is no documented evidence that the original plant design basis reliability as stated in the FSAR have been satisfied.

In the original design, separate contact output switches on the pressure and level sensors were mechanically linked by the sensor internal mechanism. These individual switches were analyzed for postulated failure mode and effects (Reference 1). In the transmitter design modification, new failure modes have been introduced by the addition of components required for the transmitter instrument loop. There is no documented evidence that these postulated failure modes have been considered, and that their effects on the involved safety-related systems have been assessed. Such analyses are required by IEEE Std. 352-1975, which was listed as an applicable design input (standard) for this modification. ANSI N45.2.11 requires that design activities be accomplished in a manner which assures that applicable design inputs are correctly translated into design output.

Response time values for the original pressure and level sensors were provided in NEDO 10139 (Reference 1) and the Dresden 3 FSAR Chapter 7

Tables. These values were used in the transient and accident analyses provided in Chapter 14 of the FSAR. There is no documented evidence that response time differences introduced by the transmitter design modification have been identified and evaluated relative to the design basis event analyses. This violates a design basis requirement stated in IEEE Std. 279-1968 (Reference 9).

REFERENCES


1. General Electric Topical Report NEDO 10139, "Compliance of Protection Systems to Industry Criteria," June 1970.
2. Dresden 3 Design Modification M12-3-83-36, Core Spray Flow Switches to Flow Transmitters FT-3-1464A and B.
3. Dresden 3 Design Modification M12-3-83-37, Pressure Suppression Differential Pressure Switches to Differential Pressure Transmitters DPT-3-1622A and B and Level Switch to Level Transmitter LT-3-1626.
4. Dresden 3 Design Modification M12-3-83-38, High Pressure Coolant Injection Pressure and Differential Pressure Switches to Pressure Transmitters PT-3-2389A through 3-2389D and Differential Pressure Transmitters DPT-3-2352 and 3-2353.
5. Dresden 3 Design Modification M12-3-83-39, RHR Flow and Differential Pressure Switches to Flow Transmitters FT-3-1501-58A and B and Differential Pressure Transmitters DPT-3-1543A and B.
6. Dresden 3 Design Modification M12-3-83-40, Reactor Vessel Water Level Switches to Level Transmitters LT-3-263-57A and B, LT-3-263-58A and B, and LT-3-263-73A and B.
7. Nutech Loop Schematic/Functional Block Diagram - Analog Trip System Class 1E Instrumentation Upgrade, 12E-7820, Rev. E, 8/22/85.
8. CECo Modification Approval Letter, M12-3-83-36 through -40, 6/6/84.
9. IEEE Std 279-1968, Criteria for Protection Systems for Nuclear Power Generating Stations.

U4.1-3 (Unresolved Item) Selection of Interruptible Power Sources for Analog Transmitter Replacement Modifications

DESCRIPTION: The original BWR design used direct acting pressure and differential pressure switches for automatic initiation of reactor protection, containment isolation, and emergency core cooling systems. For the operational Dresden and Quad-Cities Stations, Nutech engineers developed the design modification to convert certain sensor switches to analog transmitters.

In the original Dresden 3 design, fail-safe reactor protection and some containment isolation sensors were powered from the reactor protection system motor-generator sets. Emergency core cooling and the remaining containment isolation sensors were powered from 125 volt dc battery sources. These design basis power source selections met the vital instrumentation power requirements as stated in IEEE Std. 308-1971 and 1974 versions (Reference 7) even though a specific FSAR commitment to this standard has not been identified by the team.

Approximately half of the sensor channels were placed on the 24 volt dc neutron monitoring system batteries, and the other half were placed on interruptible ac busses derived from offsite power sources backed up by the onsite diesel generators (References 8 and 9). The team was concerned with the technical adequacy of the interruptible power source selections since they do not conform with the original Dresden 3 design basis.

In addition the team was concerned that the change may have increased the probability of a malfunction of the reactor protection and engineered safety features systems. The team determined that the licensee's 10 CFR 50.59 safety analysis for this modification was deficient (see Deficiency D4.1-7).

The team reviewed the Nutech power source selection for each of the 21 transmitter substitutions specified for Dresden 3 (References 2 through 6), and determined with Nutech personnel that the selection of interruptible power sources appears to have the following effects:

(1) Reactor protection system channels that normally have a 60 second delay before motor-generator power is lost would now immediately initiate a "half-scram" condition during the 10 second interval between loss of offsite power and restoration of ac power by the diesel generators. Reactor vessel water level channels B1 and B2 would be affected (Reference 9). Manual reset would be required to remove the "half-scram" condition after power is restored.

(2) Containment isolation signals have the same characteristic as the reactor protection system, and would immediately initiate a "containment half-isolation" condition. Manual reset would be required after power is restored.

(3) Some engineered safety feature sensors that would have been unaffected by the loss of offsite power due to their connection to 125 volt dc battery sources are now dependent on restoration of interruptible ac power. Torus vacuum, core spray high flow,

reactor vessel water level permissive for containment spray, and residual heat removal heat exchanger differential pressure sensors would be affected (References 8 and 9). Manual reset is not required to restore these engineered safety feature systems to their normal readiness state; therefore, the sensors would initiate any required safety actions after restoration of ac power is accomplished.

In summary, several manual reset actions are now required to return the affected systems to their normal state after power is restored which increases the probability of spurious reactor trip and containment isolation. The use of interruptible power sources increases the sensor channel contribution to composite engineered safety feature system unavailability, and results in a lower availability for the affected systems (Reference 10).

BASIS: The Nutech choice of interruptible power sources for approximately half of the transmitter substitutions does not conform to the original plant design basis for Dresden 3 (Reference 1). Technical adequacy of the interruptible power source substitution was not analyzed or justified.

10 CFR 50.59 allows licensees to make changes to the facility as described in the FSAR without prior NRC approval, unless an unreviewed safety question is involved. A change which increases the probability of malfunction of equipment important to safety is an unreviewed safety question. The licensee's safety evaluation did not adequately assess potential increased probability of equipment malfunction associated with this modification.

REFERENCES

1. General Electric Topical Report NEDO 10139, "Compliance of Protection Systems to Industry Criteria," June 1970.
2. Dresden 3 Design Modification M12-3-83-36, Core Spray Flow Switches to Flow Transmitters FT-3-1464A and B.
3. Dresden 3 Design Modification M12-3-83-37, Pressure Suppression Differential Pressure Switches to Differential Pressure Transmitters DPT-3-1622A and B and Level Switch to Level Transmitter LT-3-1626.
4. Dresden 3 Design Modification M12-3-83-38, High Pressure Coolant Injection Pressure and Differential Pressure Switches to Pressure Transmitters PT-3-2389A through 3-2389D and Differential Pressure Transmitters DPT-3-2352 and 3-2353.
5. Dresden 3 Design Modification M12-3-83-39, RHR Flow and Differential Pressure Switches to Flow Transmitters FT-3-1501-58A and B and Differential Pressure Transmitters DPT-3-1543A and B.
6. Dresden 3 Design Modification M12-3-83-40, Reactor Vessel Water Level Switches to Level Transmitters LT-3-263-57A and B, LT-3-263-58A and B, and LT-3-263-73A and B.
7. IEEE Std. 308-1974, Class 1E Power Systems for Nuclear Power Generating Stations, Section 5.4.
8. Nutech Wiring Diagram 12E-7861, Rev. D, 8/19/85.
9. Nutech Wiring Diagram 12E-7860, Rev. D, 10/7/85.
10. Dresden 3 Updated Safety Analysis Report, Table 6.2.7.4, "ECCS Availability -- Large Break with Auxiliary Power."

U4.1-4 (Unresolved Item) High and Low Pressure Coolant Injection (HPCI and LPCI) System Dependency on Neutron Monitoring System Battery Sources for Analog Transmitter Replacement Modifications

DESCRIPTION: The original BWR design used direct acting pressure and differential pressure switches for: (1) automatic termination of HPCI based on low steam pressure measurements at the HPCI turbine; (2) automatic termination of HPCI based on high steam line flow measurements that are indicative of a steam line break, and (3) automatic closing and subsequent reopening of LPCI minimum flow bypass valves based on flow measurements. Output contacts of these particular switches were energized from the safety-related 125 volt dc batteries used for engineered safety feature sensor, logic, and equipment actuation circuits. Dresden 3 and other BWR plants have replaced some pressure and differential pressure switches with electronic transmitters instrument loops.

Nutech engineers developed the design modifications to convert these emergency core cooling system sensor switches to analog transmitters (References 1 and 2). Selection of 24 volt dc battery power sources, rather than 125 volt dc, was made to provide direct voltage compatibility with the analog transmitter trip units. The team noted that an added dependency on 24 volt dc neutron monitoring system battery sources was introduced by this design modification. There was no documented evidence that the effect of this change on portions of both the high and low pressure core cooling systems had been assessed.

Based on the team's discussions with Nutech personnel regarding the transmitter design modification, postulated loss of the 24 volt dc neutron monitoring system batteries would: (1) prevent closing of the LPCI minimum flow bypass valve when required for emergency core cooling, and (2) prevent automatic termination of HPCI for either a steam line break or when low steam pressure is reached.

In addition the team was concerned that the change may have increased the probability of a malfunction of these engineered safety features systems. The team determined that the licensee's 10 CFR 50.59 safety analysis for this modification was deficient (see Deficiency D4.1-7).

BASIS: An additional system interdependency among HPCI, LPCI, and the neutron monitoring system batteries has been introduced by the design modification, without an assessment of its impact on emergency core cooling system reliability and performance.

10 CFR 50.59 allows licensees to make changes to the facility as described in the FSAR without prior NRC approval, unless an unreviewed safety question is involved. A change which increases the probability of malfunction of equipment important to safety is an unreviewed safety question. The licensee's safety evaluation did not adequately assess potential increased probability of equipment malfunction associated with this modification.

REFERENCES

1. Dresden 3 Design Modification M12-3-83-38, High Pressure Coolant Injection Pressure and Differential Pressure Switches to Pressure Transmitters PT-3-2389A through 3-2389D and Differential Pressure Transmitters DPT-3-2352 and 3-2353.
2. Dresden 3 Design Modification M12-3-83-39, RHR Flow and Differential Pressure Switches to Flow Transmitters FT-3-1501-58A and B and Differential Pressure Transmitters DPT-3-1543A and B.
3. USNRC Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Rev. 3, November 1978, Sections 6.2.1.7, 7.2.2, and 7.3.2.

U4.1-5 (Unresolved Item) Environmental Qualification Documentation for Analog Transmitter Replacement Modifications

DESCRIPTION: Nutech engineers developed the design modifications to convert certain reactor protection system and emergency core cooling system initiation sensor switches to analog transmitters. (References 1 through 5). For the replacement transmitters and related components, the team compared environmental qualification requirements with the qualification test plan (References 6 through 8).

Some of the required environmental qualification values do not appear to have been satisfied by the qualification test plan. These values and a number of design document inconsistencies for environmental qualification are as follows:

(a) Agastat catalog data for the master and slave relays in the Nutherm qualification procedure (Reference 7) stated that 125 volt dc relays were tested; however, the Rosemount trip unit output (Reference 8) required that 24 volt dc relays be used.

(b) The Nutherm EQ procedure (Reference 7) stated the abnormal temperature value as 104°F, but the Nutherm installation, operation, and maintenance manual (Reference 6) listed the value as 120°F. The test program used 104 plus 15 degrees for margin.

(c) The Nutherm EQ procedure did not list the 24 volt dc input used for some of the channels (References 1 through 5). There was no indication that the test used 24 volts dc.

(d) The Nutherm EQ procedure stated normal and abnormal humidity as not specified and less than 100% respectively; however, the Nutherm manual (Reference 6) listed values of 80 and 90 percent respectively.

(e) The Nutherm EQ procedure stated gamma radiation as negligible; however, the Nutherm manual (Reference 6) listed a value of 1.1 E+6 RADS integrated dose.

(f) The Nutherm EQ procedure had an acceptance criteria that permits up to 2 milliseconds of relay contact chatter; however, there was no documented evidence that such spurious relay contact operation was acceptable for the particular trip channels involved in this modification.

BASIS: IEEE Std. 323-1974 Section 5 states that the preferred qualification method is a test of actual equipment using simulated service conditions. This standard requires that the capability of all Class 1E equipment for performing its required function be demonstrated; that assurance be provided that the severity of the qualification methods equal or exceed the maximum

anticipated service requirements and conditions; and that qualification documentation contain the performance requirements, the methods used, results, and justifications. The team did not find evidence that design documentation inconsistencies pertinent to the anticipated service conditions for this equipment had been identified and resolved in the qualification process.

REFERENCES


1. Dresden 3 Design Modification M12-3-83-36, Core Spray Flow Switches to Flow Transmitters FT-3-1464A and B.
2. Dresden 3 Design Modification M12-3-83-37, Pressure Suppression Differential Pressure Switches to Differential Pressure Transmitters DPT-3-1622A and B and Level Switch to Level Transmitter LT-3-1626.
3. Dresden 3 Design Modification M12-3-83-38, High Pressure Coolant Injection Pressure and Differential Pressure Switches to Pressure Transmitters PT-3-2389A through 3-2389D and Differential Pressure Transmitters DPT-3-2352 and 3-2353.
4. Dresden 3 Design Modification M12-3-83-39, RHR Flow and Differential Pressure Switches to Flow Transmitters FT-3-1501-58A and B and Differential Pressure Transmitters DPT-3-1543A and B.
5. Dresden 3 Design Modification M12-3-83-40, Reactor Vessel Water Level Switches to Level Transmitters LT-3-263-57A and B, LT-3-263-58A and B, and LT-3-263-73A and B.
6. Nutherm Installation, Operation, and Maintenance Manual, C-1167-M1, Rev. 0, 1/9/84, pages 1, 6, 24, and 26.
7. Nutherm Equipment Environmental Qualification Procedure, N-128-001P, Rev. 0, 10/26/83, pages 4, 6, 10, and 23.
8. Rosemount Trip/Calibration System Model 710DU Instruction Manual, Rev. A, 4/83, page 7.

U4.1-6 (Unresolved Item) Common Instrument Line to Redundant Instruments for Analog Transmitter Replacement Modifications

DESCRIPTION: The original Dresden 3 design had redundant torus pressure switches DPS-3-1622A and B connected to a common instrument line with an in-line valve (Reference 2). The safety function of these pressure switches is to sense a vacuum condition in the torus and open the reactor building to suppression chamber vacuum breakers to relieve the vacuum condition. Undetected closure of the in-line valve or plugging of the instrument line would prevent the pressure switches from performing this intended safety function.

As required by GE Specification 22A2501 (Reference 1), the physical connection of process sensors should be coordinated with the safety system coincidence logic to assure compliance with single failure and separation criteria (Reference 1). In the transmitter replacement modification for these switches, the postulated single failure of the instrument line was not identified (Reference 3).

BASIS: For the transmitter design modification involving redundant torus pressure measurement channels, a common instrument line and in-line valve were not identified as a violation of the single failure criterion described in applicable separation criteria (Reference 1) and IEEE Proposed Standard 279-1968.

REFERENCES

1. General Electric Specification, 22A2501, "Separation Requirements for Reactor Safety and Engineered Safeguards Systems," Rev. 0, 1/28/69.
2. Sargent and Lundy Piping and Instrument Diagram, M-356, Pressure Suppression Piping, Rev. AE, 7/5/85.
3. Dresden 3 Design Modification M12-3-83-37, Pressure Suppression Differential Pressure Switches to Differential Pressure Transmitters DPT-3-1622A and B.

4.1-7 (Deficiency) Failure to Adequately Document Safety Evaluations for Plant Modifications

DESCRIPTION: 10CFR50.59 requires that the licensee prepare a written safety evaluation which provides the basis for the determination that a facility change does not involve an unreviewed safety question. The team noted that, for the modifications addressed in this inspection, the safety evaluations prepared by SNED tended to be brief, containing no detailed bases for the scenarios evaluated or for the conclusions reached. There was no documented evidence that the safety evaluations for modifications M12-3-83-36 through -40 (analog trip system) considered the impact of changed system response times, a new system interface between the analog trip system and several power supplies, and the loss of channel separation.

BASIS: This a violation of the requirements of 10CFR50.59.

REFERENCES: None.

D4.2-1 (Deficiency) Setpoint Calculations for Safety-Related Instruments

DESCRIPTION: The team reviewed a Nutech letter (Reference 4) that delineated instrument loop current setpoint values for analog transmitters which replaced 21 direct acting pressure and level switches (Reference 1). Dresden 3 Station personnel developed data for process measurement ranges and current setpoints, and Nutech converted the analog trip unit setpoints to milliampere values (Reference 4). The Nutech letter indicated that a number of setpoint values could not be correlated to system level design basis requirements, such as a stated percent of rated system flow for line break detection.

Because transmitters exhibit different performance characteristics than direct acting switches, the team reviewed available site documentation for aspects such as range, instrument loop accuracy, setpoint value, setpoint tolerance, response time, and dead-band.

Dresden 3 Station personnel were maintaining documented records of safety-related setpoint values, but setpoint margin was being added at the site without a documented basis. No procedure exists to govern the calculation of safety-related setpoint values. CECo Procedure DAP-11-11 (Reference 2) required that setpoint changes be supported by a safety impact evaluation. Such an evaluation had not been performed for the two setpoint changes reviewed by the team.

BASIS: Section 3 of IEEE Proposed Standard 279-1968 requires a documented design basis for safety-related instrument setpoint values. There was no evidence of appropriate Dresden 3 setpoint documentation for the transmitter substitution modifications, even though this standard was referenced in the modification approval letter.

In addition, safety impact evaluation requirements of Commonwealth Edison procedures were not implemented for two calculations reviewed by the team.

REFERENCES

1. CECo Modification Approval Letter, M12-3-83-36 through -40, Analog Transmitter Design Modification, 6/6/84.
2. CECo Procedure DAP-11-11, Rev. 6, 7/85, page 5.
3. CECo Procedure DAP-5-1, Rev. 13, 10/85.
4. Nutech Letter XCE-04-363, Setpoint Data for EQ Instrumentation Replacement, 5/9/85.

D4.3-1 (Deficiency) Component Classification Errors

DESCRIPTION: For the Dresden 3 Station, Commonwealth Edison maintains a master equipment list that contains summary information regarding components in safety-related systems (Reference 1). This list was compiled by Sargent and Lundy from design documents, analyses, reports, and other sources developed by a number of design firms associated with the plant (Reference 9). In particular, the safety class designation of mechanical and structural components was provided by Sargent and Lundy, electrical components by Nutech, and environmentally qualified equipment by Bechtel. The team reviewed a number of individual components included in the master equipment list for their safety class designation, and identified design process control errors with respect to (1) incorrect Nutech input, (2) incorrect use by Sargent & Lundy of the Nutech input in compiling the master list, and (3) non-compliance by Nutech with its report control instructions.

One input to the master equipment list was a safety-related electric equipment list report prepared by Nutech (Reference 2). Electrical devices excluded from the Nutech list were stated to be non-safety-related based on a brief failure mode and effects analysis. The team determined that the following types of reactor protection system and LPCI loop selection valve control components had been incorrectly designated as non-safety-related:

(1) Manual switches and relays used for reactor manual trip;

(2) Sensors and relays used for anticipatory reactor trips;

(3) A manual switch and relays used for reset of reactor trip;

(4) Fuses used in normally energized reactor trip circuits;

(5) Control switches for the recirculation pump suction valve, and;

(6) Control switches and relays for the recirculation pump discharge valve.

In FSAR Amendment 22 page 2-3, the results of a BWR common mode failure analysis are described (Reference 6). With respect to manual reactor trip, this report stated that "in the presence of a common-mode failure, prompt and proper operator action is assumed as needed." Similarly, this report identified a number of anticipatory trips that provide a diverse backup to postulated common mode failures in other portions of the reactor protection system. Another report (Reference 7) provided detailed compliance analyses of the generic BWR protection system to IEEE Proposed Std. 279-1968. This established design basis for reactor protection system components indicates that the above components are safety-related.

Despite the Nutech non-safety-related designation for these components, the compiled master equipment list did identify each component reviewed by the

team as safety-related. Commonwealth Edison instructions to Sargent and Lundy indicated that all components identified by Nutech were to be included in the compiled list; hence, those components designated by Nutech as non-safety-related were actually included as safety-related components on the master equipment list without regard to the Nutech classification designation.

Each version of the Nutech electric equipment list did not identify the report either as "for information only" or provide the alternative preparer, reviewer, and approver signatures, as required by Nutech Procedure QEP 3-3 (Reference 8). This procedure, QEP 3-3, requires that "reports not used as a basis for design should state that they are for information only." The hardcopy report reviewed by the team had equipment tabulation sheets dated 11/23/84 and was designated as Revision 1, but the report was not signed, dated, or numbered. A later report version prepared from computer diskettes was also designated as Revision 1, but had different text material and its tabulated sheets were dated 3/8/85. Even though this work involved the identification of safety-related components, there was no indication that the electrical equipment list was internally reviewed for accuracy and correctness by Nutech.

BASIS: Dresden 3 FSAR figures (References 3 and 4) identify that the manual trip, anticipatory trip, and fuse components are part of the reactor protection system. These figures are consistent with licensing commitments (Reference 5) and the reactor protection system scope described in IEEE Proposed Std. 279-1968 and NUREG-0800 Branch Technical Position ICSB 26 (Reference 10). Contrary to these requirements, the Nutech electric equipment list incorrectly classified certain reactor protection system components as non-safety-related. Manual controls and relays used for recirculation pump suction and discharge valve control were also incorrectly classified as non-safety-related.

The Nutech report was not issued and controlled in accordance with the applicable Nutech Procedure QEP 3-3.

REFERENCES

1. Dresden 2 and 3 Master Equipment List (Mechanical/Electrical), Rev. 1, 9/30/85.
2. Nutech Engineers Report, "CECo Dresden 3 Safety-Related Electrical Equipment List," Rev. 1, 9/23/84 (hardcopy) and Rev. 1, 3/8/85 (diskette).
3. FSAR Figure 7.7.1, Reactor Protection System - Single Logic Channel Tripping Diagram, Rev. 3, 3/22/68.
4. FSAR Figure 7.7.2, Reactor Protection System Scram Function, Rev. 3, 3/22/68.
5. FSAR Amendment 11 Questions II.B.2 and II.B.3.
6. General Electric Topical Report, NEDO-10189, "An Analysis of Functional Common Mode Failures in GE BWR Protection and Control Instrumentation," June 1970, page 21, Table 3, and page 4.

7. General Electric Topical Report, NEDO-10139, "Compliance of Protection Systems to Industry Criteria," June 1970, Sections 2.4, 2.5, 2.2.12, 2.2.19, and Figure 2-111.
8. Nutech Quality Engineering Procedure 3-3, Rev. 4, 8/30/85.
9. CECo Action Item Record, AIR-12-84-130, Master Equipment List Development, 6/11/84.
10. NUREG 0800, Branch Technical Position ICSB 26, "Requirements for Reactor Protection System Anticipatory Trips," Rev. 2, 7/81.

D4.3-2 (Deficiency) Safety Classification of Master Equipment List Motor Components

DESCRIPTION: Commonwealth Edison maintains a master equipment list for Dresden 3 that describes the safety classification of each component. The team identified the following safety-related motor components which were incorrectly classified as non-safety-related:

(a) Actuator motor for MOV-MO-2301-3; the steam supply valve to the HPCI turbine. Failure of this valve to open on a HPCI initiation signal will prohibit HPCI system operation when demanded.

(b) Actuator motors for MOV-MO-2301-36 and -06; the HPCI pump suction valves from the suppression pool and condensate storage tank respectively. Failure of these valves to change position on low water level in the condensate storage tank will terminate operation of the HPCI system when low water level in the tank is reached.

(c) Actuator motors for MOV-MO-2301-48 and -49; the HPCI pump cooling leakage water return line valves to the condensate storage tank. During HPCI operation with pump suction taken from the suppression pool, failure of these valves to close may result in contaminated water being introduced into the condensate storage tank located outside of the reactor building.

(d) Pump motors for 1501-044A, -B, -C, and -D pumps. These four pumps provide service water to the containment cooling heat exchangers in the LPCI system. Failure of the motors would disable the pumps and prevent the containment cooling heat exchanger from performing its intended function.

BASIS: 10CFR Part 100, Appendix A states that structures, systems, and components which must remain functional following a safe shutdown earthquake include those necessary to assure: (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of this part. This subset of items has been accepted by the NRC as defining "safety-related" structures, systems, and components. The above listed motor components are necessary to assure achievement of items (2) and (3) above.

REFERENCES

1. Master Equipment List, Dresden Unit 3, 9/85.
2. Drawing M-374, HPCI System Piping and Instrument Diagram, Rev. AH.
3. Drawing M-360, LPCI System Piping and Instrument Diagram, Sheet 1, Rev. TU and Sheet 2, Rev. F.

D4.4-1 (Deficiency) Reactor Water Clean-Up Room Leak Detection System Safety-Related and Non-Safety-Related Circuit Isolation

DESCRIPTION: Two non-safety-related temperature monitors were added to the equipment room containing portions of the reactor water cleanup system. The purpose of this change was to detect leakage from a postulated crack in Safety Class 1 piping and provide an alarm to the control room operator (Reference 2).

The schematic diagram for this design modification (Reference 1) showed that each monitor was connected to a safety-related Class 1E power source without adequate isolation, such as by use of double circuit breakers. In a letter addressing Systematic Evaluation Program concerns dated December 6, 1982, Commonwealth Edison committed to provide either a short circuit analysis or a breaker coordination study for the connection of non-Class 1E loads to Class 1E busses. However, the team was unable to locate either the short circuit analysis or the breaker coordination study for this particular modification to confirm that Class 1E busses were not degraded below an acceptable level by the connection of non-Class 1E loads.

After the team identified this omission, Commonwealth Edison performed a breaker coordination study that confirmed the adequacy of these Class 1E connections to non-Class 1E loads.

BASIS: Analyses to support the connection of non-Class 1E loads to Class 1E busses were not prepared in accordance with a Systematic Evaluation Program commitment (Reference 3).

REFERENCES

1. Drawing 12E-7862, RWCU Leak Detection System, Rev. A.
2. Design Modification M12-3-84-17, RWCU Leak Detection.
3. Dresden 3 Systematic Evaluation Program Report NUREG-0823, Section 4.21.5.

U4.5-1 (Unresolved Item) Rod Worth Minimizer Design Basis Requirements

DESCRIPTION: The design basis for the original Dresden 3 rod worth minimizer system was provided in a General Electric topical report (Reference 1). This system imposes control rod withdrawal and insertion blocks to preclude the development of an inadvertent high rod worth condition. Two Dresden 3 design modifications (References 2 and 3) involve replacing the existing rod worth minimizer input buffer, computer system, and output buffer components with solid-state equipment. In the technical description of the proposed rod worth minimizer equipment upgrade, one design basis requirement appears to have been overlooked and a second requirement was not explicitly stated in the RWM-1 specification (Reference 4):

(1) In the previous design, an analog "dead-man switch" circuit imposed rod movement blocks when the rod worth minimizer equipment was inoperative. In the proposed design, the operator is provided with an alarm, but rod blocks are not directly imposed.

(2) The rod worth minimizer was originally designed to be "fail-safe" with respect to loss of power and impose rod movement blocks until the keylock bypass switch was placed in the bypass position. This requirement was not explicitly stated in the RWM-1 specification; however, the proposed design does describe this fail-safe characteristic in the process computer system's hardware specification.

BASIS: The proposed rod worth minimizer design modifications did not comply with one original design basis requirement for a rod movement block as stated on page 7.9.3.3. of the Dresden 3 Updated Safety Analysis Report, and no justification was developed for its omission. A second design basis described in the General Electric Topical Report was not explicitly stated in the RWM-1 technical document that provided specific performance requirements for the modification.

REFERENCES

1. General Electric Topical Report, APED-5449, Control Rod Worth Minimizer, March, 1967.
2. M12-3-84-4, Rod Worth Minimizer Design Modification, 1/18/84.
3. M12-3-84-6, Rod Position Indication System Output Buffer Design Modification, 10/21/85.
4. Rod Worth Minimizer Specification, RWM-1, Rev. 1, undated.

D5.1-1 (Deficiency) 250 Volt Battery Sizing Calculation

DESCRIPTION: Dresden Units 2 and 3 both utilize a 250 volt dc system to supply power to large direct current loads such as pumps, motor operated valves, and instrument inverters. Each unit has one battery which supplies power to its own Division I loads and the Division II loads of the opposite unit. Sargent & Lundy provided the engineering associated with replacing the original batteries with new larger batteries, consisting of fewer but larger cells of a different type than the original cells.

Sargent & Lundy prepared specification T-3349 (Reference 1) to purchase the new batteries. The loading requirements for the new batteries were based upon Sargent & Lundy calculation 705600-19-4 (Reference 2). The team reviewed this calculation and had the following concerns:

1. Motor data for the dc auxiliary pumps and motor operated valves were taken from the National Electrical Code instead of using actual plant specific motor data. No reference was provided for the source of data for those motor horsepowers not given in the National Electrical Code, nor was there justification for the use of other than plant specific motor data.

2. The calculation referenced IEEE 485, an accepted industry standard, as the method for battery sizing. Section 4.2.3 of IEEE 485 states that momentary loads, such as motor starting currents, should be considered. However, motor starting currents were neglected without justification for both pump and motor operated valve loads. While the one line drawing (Reference 3) indicates that the pump motor starting currents are controlled with multiple step starters, no reference was supplied to indicate the limiting value of the starting current. Motor operated valves do not utilize stepping starters.

3. Cell sizing calculation 705600-19-6 Rev.0, 2/28/85 (Reference 4) used a temperature correction factor for 70°F without justification. The purchasing specification for the batteries and the cell sizing calculation for the 125 volt battery (Reference 5) both agree with the CECo System Standard N2 (Reference 6) which states that Class 1E batteries should operate over a normal temperature range down to 65°F. The present plant surveillance procedures (Reference 7) do not require the battery to be declared inoperable until the cell temperature drops below 60°F.

BASIS: ANSI N45.2.11 (Reference 8) Section 6.3.1 requires that design input be correctly selected and incorporated into design, assumptions be adequately described and reasonable, and correct design methods be used. Sargent & Lundy Procedure GQ-3.08 (Reference 9) requires that assumptions that must be verified should be identified in the calculation. Contrary to these requirements, Sargent & Lundy failed to evaluate the effect that actual Dresden motor data and motor starting currents would have

on the sizing calculation. The assumed minimum battery room temperature was inconsistent with other sources of this parameter. The referenced IEEE 485 was not correctly implemented.

REFERENCES


1. S&L Specification T-3349, Storage Batteries, 8/7/85.
2. S&L Calculation 705600-19-4, 250 Volt Battery Sizing, 2/8/85.
3. S&L Dwg. 12E-2321 Rev. N, 250 Volt Bus Arrangement, 9/28/84.
4. S&L Calculation 705600-19-6, 250 Volt Cell Sizing, 2/28/85.
5. S&L Calculation 705600-19-9, 125 Volt Battery Sizing, 1/31/85.
6. CECo System Standard N-2, Requirements for Class 1E Nuclear Station Batteries, 3/3/83.
7. CECo Procedure DOS-6900-1, Rev. 8, Daily Storage Battery Checks, 3/85.
8. ANSI N45.2.11, Quality Assurance Requirements for the Design of Nuclear Power Plants, 1974.
9. S&L Procedure GQ-3.08, Rev. 5, Design Calculations.

D5.1-2 (Deficiency) 125 Volt Battery Cell Selection

DESCRIPTION: The original 125 volt batteries on both Dresden units were found to be too small (Reference 1) to carry the design load for the entire discharge time. In addition, because of physical cell deterioration in the Unit 2 125 volt battery, CECo decided to replace both batteries in Units 2 and 3 (Reference 2) with batteries having a different type cell. Sargent & Lundy prepared a load analysis for the 125 Volt dc system (Reference 3) and, based upon the load profile, performed a battery sizing calculation in accordance with IEEE Standard 485 (References 4 and 5). The calculation resulted in a requirement for an NCX cell with 8.1 positive plates for Unit 3 and 8.4 positive plates for Unit 2. The IEEE Standard referenced in the calculation calls for the fractional plate in both cases to be rounded up for a total of 9 positive plates. However, an NCX cell with 9 positive plates would result in a battery with a higher available short circuit current than the Dresden dc switchgear design rating.

Contrary to the IEEE Standard, Sargent & Lundy rounded the number of required positive plates downward (to 8.0), instead of modifying the switchgear or increasing the number of cells to permit a smaller cell size. This resulted in a smaller size cell than called for by the IEEE Standard. However, in the conclusion to the calculation, the preparer incorrectly stated that the battery (with 8 positive plates) was sufficiently large to carry the design load without load shedding.

BASIS: ANSI N45.2.11 (Reference 6) Section 6.3.1 requires that verifiers assure that appropriate assumptions and correct design methods are used. Sargent & Lundy Procedure GQ-3.08 (Reference 7) requires that the reviewer verify the assumptions used in a calculation. Contrary to these requirements, neither the preparer nor the verifier identified any exception to the IEEE 485 battery sizing method.

REFERENCES

1. S&L Calculation 705600-19-7, Rev. 0,125V DC (Existing) Battery Size, Open Rev.
2. CECo Modification Package No. M12-2(3)85-25, 125 V Battery Replacement.
3. S&L Calculation 705600-19-5 Rev. 1, 125 Volt Load Estimation, Open Rev.
4. S&L Calculation 705600-19-9 Rev. 0, 125 Volt Cell Selection, Open Rev.
5. IEEE-485-1983, Recommended Practice for Sizing Large Lead Storage Batteries.
6. ANSI N45.2.11, Quality Assurance Requirements for the Design of Nuclear Power Plants, 1974.
7. S&L Procedure GQ-3.08, Rev. 5, Design Calculations.

05.1-3 (Observation) 125 Volt Battery Sizing Calculation

DESCRIPTION: Dresden Units 2 and 3 both utilize a 125 Volt dc system to supply power to switchgear control and other low power direct current loads (Reference 1). Each unit has one battery which supplies power to its own Division I loads and the Division II loads of the opposite unit. Sargent & Lundy provided the engineering associated with replacing the original batteries with new larger batteries.

S&L prepared specification T-3349 (Reference 2) to purchase the new batteries. The loading requirements for the new batteries were developed by calculation 70560019-5 (Reference 3). This calculation being reviewed and verified by off-project S&L personnel. Calculation sign off was required before making a recommendation to purchase the new batteries, and had not yet occurred. The team reviewed this calculation and identified the following concerns:

1. The calculation did not consider any dc power requirements for control of circuit breakers for 480 Volt buses 20 and 30, shown in the key oneline (Reference 5).

2. The calculation did not contain any justification for using an assumed diversity factor of 2/3 for connected loads on individual circuits throughout the calculation, including such loads as evacuation sirens and scram solenoids.

3. The load for Generator/Transformer Backup Relay Panel had not been included in the calculation.

4. The normal load for the HPCI control panel was not included in the calculation.

5. There was an inconsistency in the number of loads tripping and closing on 480 Volt Bus 28.

The team does not expect resolution of these concerns to substantially affect the overall size of the battery profile.

REFERENCES

1. S&L Dwg. 12E-3322B, Rev. A,Overall Key Diagram 125V DC Distribution.
2. S&L Specification T-3349, Addendum 1, Storage Batteries & Accessories, 8/7/85.
3. S&L Calculation 70560019-5, Rev. 0, 125 V DC Bus Load Profile, 2/28/85.
4. S&L Calculation 70660019-5, Rev. 1, 125 V DC Bus Load Profile, 11/27/85.
5. S&L Dwg 12E-2322A, Rev. C, Overall Key Diagram 125 V DC Distribution.

D5.1-4 (Deficiency) Unit 1 HPCI Battery Tie Voltage Drop Calculation

DESCRIPTION: Dresden Units 2 and 3 share two 125 volt batteries with one battery located in each unit (Reference 1). Each battery normally supplies Division I loads for the host unit and Division II loads for the opposite unit. The Dresden 2 battery (which normally supplies Division II loads in Unit 3) was observed to have signs of early failure. CECo decided to provide a temporary feed between the Unit 1 HPCI battery and Unit 2 until the Unit 2 battery could be replaced (Reference 2).

Sargent & Lundy analyzed the capability of the HPCI battery to supply the loads in Unit 2 (and the Division II loads of Unit 3). As part of this analysis, Sargent & Lundy developed a calculation (Reference 3) to check the voltage which would result at the Unit 2 dc switchgear. The team determined that the analysis was based upon preliminary load current taken from an earlier battery load analysis (Reference 4). Based upon this input, the voltage drop calculation was approved and the design modification was issued to the station without noting that the preliminary load current used in the calculation must be verified. The latest battery sizing calculation (used to size the replacement 125 V batteries) (Reference 5) develops higher load currents than used to determine the cable voltage drop pertinent to the Unit 2 dc switchgear.

BASIS: ANSI N45.2.11 (Reference 6) Section 4.2 requires that assumptions that must be verified be identified. Sargent & Lundy Procedure GQ-3.08 (Reference 7) requires that the reviewer verify that those assumptions which must be verified are identified. Contrary to these requirements, Sargent & Lundy failed to identify that the battery load currents used in the voltage drop calculations (Reference 3) were based upon a load profile that had not been verified.

REFERENCES

1. S&L Dwg. 12E-3322B, Rev. A, Overall Key Diagram, 125 V DC Distribution Center, 7/13/84.
2. CECo Modification M12-2-85-31, Temporary DC Feed to Unit 2.
3. S&L Calculation, 717500-19-1, Rev. 0, 125 V Dresden 2 Battery Replacement, 3/19/85.
4. S&L Calculation, 705600-19-7, 125 V DC Battery Sizing, 1/18/85 (Not Approved).
5. S&L Calculation, 717500-19-5, Rev. 1, 125 VC Modified DC Bus Load Profile, 11/27/85 (Not Approved).
6. ANSI N45.2.11, Quality Assurance Requirements for the Design of Nuclear Power Plants, 1974.
7. S&L Procedure GQ-3.08, Rev. 5, Design Calculations, 1/31/85.

D5.1-5 (Deficiency) Errors in Updated Documentation for 24/48 Volt Batteries

DESCRIPTION: Modification M12-2(3)-83-14 (Reference 1) was developed to replace the existing 24/48 volt batteries for Dresden Units 2 and 3. The batteries power the source range and intermediate range neutron monitors, the process radiation monitoring system, and the analog trip system. The original batteries for Units 2 and 3 were Gould DPR-19 Plante type rated 80 ampere-hours (8 hour rate). The replacement batteries for both units are Gould 2-MCX-190 Lead Calcium type rated 190 ampere-hours (8 hour rate). The Unit 3 replacement batteries were put in service during February 1984, while the Unit 2 replacement batteries were put in service in February 1985. The Updated Safety Analysis Report, revised in June 1985, erroneously referenced both the 80 ampere hour and the 190 ampere hour batteries, and erroneously stated that the Unit 3 24/48 volt batteries consisted of 96 cells instead of the correct number of 24 cells. The correct number of cells can be determined by dividing the battery voltage (48 volts) by the cell voltage (nominal 2 volts/cell).

The modification letter (Reference 1) states that, after completion of the modification, a DCR with marked drawings should be sent to the Station Nuclear Engineering Department to incorporate record revisions. The team's review of the dc system single line drawing 12E-3324 (Reference 2) determined that, while this drawing was revised in April 1984, the ampere-hour rating of the battery was not indicated as 190 to reflect the replacement batteries.

BASIS: Contrary to ANSI N45.2.11-1974, Section 3.1, changes to design inputs were not correctly documented in drawings or the Updated Safety Analysis Report.

REFERENCES:

1. CECo Letter dated September 15, 1983, "Modification M12-2(3) 83-14, Replace 24/48 Volt Battery, Dresden Station, Unit 2 and 3."
2. Commonwealth Electric Company Drawing 12E-3324 Revision H, dated April 1, 1985.
3. ANSI N45.2.11, Quality Assurance Requirements for the Design of Nuclear Power Plants, 1974.

05.1-6 (Observation) Specification of 24/48 Volt Batteries

DESCRIPTION: Surveillance tests of the Quad Cities 24/48 volt batteries indicated that the condition of the cells was deteriorating. Because of this and the coincidental release of USNRC IE Information Notice No. 83-11 (Reference 1), identifying potential seismic problems with older lead-acid storage batteries, the Dresden plant staff authorized the replacement of the 24/48 volt batteries and battery racks by Modification M12-2(3)-83-14 (Reference 2). Safety-related batteries and racks were purchased on P.O. 278730 (Reference 3). The purchase order did not specify the required seismic response spectra for Dresden Station, but simply required that the vendor furnish the same qualification report, including seismic, as was provided for similar batteries at the Quad Cities Plant. Commonwealth Edison Company concluded that the Gould generic seismic test response spectra included in the Quad Cities Plant qualification test report (Reference 4) was adequate for the Dresden Plant, but did not document the results of that review. The team agrees with this conclusion, but recommends that the review be documented.

REFERENCES

1. USNRC IE Information Notice No. 83-11, Possible Seismic Vulnerability of Old Lead Storage Batteries, 3/14/83.
2. CECo Modification M12-2(3)-83-14, Replace 24/48 Volt Battery, 9/15/83.
3. CECo Purchase Order 278730, 24/48 Volt Batteries, 7/26/83.
4. CECo Purchase Order 266294, Environmental Qualification for Class 1E Lead-Acid Storage Batteries, Quad Cities-S0517651,2.

05.1-7 (Observation) DC Breaker Coordination

The normal arrangement of the dc distribution system for Units 2 & 3 has each unit's battery supplying its own Division I loads while at the same time supplying the Division II loads in the opposite unit (Reference 1). Provisions are included in the design which would permit each unit's battery to become the reserve supply for its own Division II loads or the opposite unit's Division I loads. This could potentially result in a single battery supplying power for both Divisions in the same unit.

In response to an Appendix R concern, CECo modified the 125 volt dc distribution system (Reference 2) to physically separate the Division I distribution equipment from the Division II equipment. As part of this modification, Sargent & Lundy performed a coordination study (Reference 3) based on the standard mode of the dc system connection (as defined in the scope of the calculation). The conclusion stated that there was no problem in the coordination of the dc system protective devices.

This conclusion statement could be misleading when the plant is connected in the reserve mode, as seen by the following example. When the Unit 3 battery is feeding the Unit 3 Division II loads, a fault on the reactor building dc bus could result in loss of both division loads. The conclusion statement should have identified the potential for lack of coordination when the system is connected in the reserve feed mode.

REFERENCES

1. S&L Dwg. 12E-3322B, Rev. A, Overall Key Diagram 125 V Distribution.
2. CECo Modification M12-2(3)-81-12, Installation of New 125 Volt DC Switchgear for Fire Protection.
3. S&L Calculation 556931-19-2, Rev. 1, 125 Volt DC System Breaker/Fuse Coordination, 11/22/85.

D5.1-8 (Deficiency) Battery Temperature Surveillance for Existing Battery

DESCRIPTION: The batteries are designed to supply the dc system with sufficient capability to maintain the plant in a safe condition following loss of the ac supply. IEEE 485 (Reference 1) provides recommendations to size lead acid cell batteries, which include accounting for cell temperatures. Temperature of the cell affects the battery's capability to deliver power. In order to maintain the battery availability and capability, Plant Technical Specification 4.9 requires battery surveillance. Cell temperature is explicitly mentioned as one item to be monitored. Plant operating surveillance procedures (References 2 and 3) require the cell temperature to be maintained above 60°F. Based on review of the battery sizing calculation (Reference 4), the team determined that the existing batteries were not sized with a temperature correction to account for cell temperature below the manufacturer's standard rating temperature of 77°F. The team also determined that a recent analysis (Reference 5) performed by Sargent and Lundy to establish the amount of load shedding required with the existing batteries also failed to include any temperature correction factor even though the analysis referenced the latest industry standard on battery sizing (IEEE 485). Operating the batteries 17°F below their rated temperatures could result in loss of battery capacity of 11%.

BASIS: Dresden Technical Specification Section 4.9, Surveillance Requirement Basis, states that the checks described in the surveillance section provide adequate indication that the batteries have the specified ampere-hour capability. The inconsistency between the battery temperature rating (77°F) and the surveillance requirement (60°F) indicates that this Technical Specification Basis cannot be met.

REFERENCES

1. IEEE-485, Recommended Practice for Sizing Large Lead Storage Batteries, 1983.
2. CECo Procedure DOS 6900-1, Rev. 8, Daily and Weekly Storage Battery Checks, 3/85.
3. CECo Procedure DOS 6900-4, Rev. 2, Quarterly Storage Battery Checks, 8/84.
4. S&L Calculation JO 3447, Analysis of Battery Requirements, 1/23/67.
5. S&L Calculation 705600-19-7, Rev. 0, 125 Volt DC Battery Sizing, 1/18/85.

D5.2-1 (Deficiency) Documentation Lacking for Cable Selection

DESCRIPTION: Modifications M12-2(3)-83-36 through 40 (Reference 1) were prepared to replace non-qualified components with environmentally qualified equipment. The original equipment consisted of process actuated switches, while the replacement equipment consists of instrumentation loops with electronic components and relays. The latter are sensitive to low voltage operation, and are to be powered in part from the unregulated 24/48 volt dc distribution system. This decision was documented in meeting notes (Reference 2) and summarized in a Nutech letter (Reference 3). Since the equipment is sensitive to low voltage operation, the cable voltage drop is a limiting condition for sizing the cable. However, the team found no calculation or other documented evidence that the cable voltage drop was considered.

BASIS: The licensee has committed to Regulatory Guide 1.64 (References 4 & 5) which endorses ANSI N45.2.11 (Reference 6). Section 3.1 of that document requires that applicable design inputs be identified, documented and controlled. The Station Nuclear Engineering Department, "Guidebook for Architect-Engineers", requires that copies of safety-related calculations be retained for a minimum of 5 years. Contrary to these requirements, no supporting calculations were maintained to substantiate this aspect of the design.

REFERENCES:


1. CECo Letter, Modification M12-2(3)-83-40, Replacement of Yarway Reactor Water Level Switches, June 6, 1984; Modification M12-2(3)-83-36 through 39, Replacement of Pressure, Differential Pressure, and Flow Switches, Dresden Station Units 2 and 3, AIR 12-82-67.
2. Nutech Communication Record, EQ Instrumentation Replacement Comment Meeting (XCE-04), 6/25/84.
3. Nutech letter XCE-04-161 to Commonwealth Edison Company - B.M. Viehl, July 23, 1984.
4. CECo Quality Assurance Manual, Quality Requirement 2.0.
5. Regulatory Guide 1.64, Quality Assurance Requirements for the Design of Nuclear Power Plants, 1974.
6. ANSI N45.2.11 Quality Assurance Requirements for the Design of Nuclear Power Plants, 1974.

05.2-2 (Observation) Maximum Voltage of Agastat Relays

DESCRIPTION: Modifications M12-2(3)-83-36 through 40 (Reference 1) were prepared to replace non-qualified components with environmentally qualified components. Several of the components are powered from the 24 V dc distribution system which has a voltage range between 28 volts, during battery equalization, and 21.7 volts at battery full discharge voltage. The voltage range of the Agastat relays used as contact output devices is 19 to 27 volts.

The team was concerned because of the problems identified in IE Information Notice No. 83-08 (Reference 2) that the relays may not be able to perform over the full range of the power supply voltage. Nutech referred to a voltage drop study (Reference 3) which determined that the relay voltage during equalization is approximately 27.3 volts. Nutech contacted the relay manufacturer, Agastat, who agreed that the relatively short duration of the equalizing voltage and the minor overvoltage of approximately 0.3 volts would not degrade the relay operation. Bechtel, who originally specified the relays, also confirmed with the relay manufacturer that the intermittent overvoltage operation of the relays would not degrade their performance (Reference 3).

REFERENCES

1. CECo letter, Modification M12-2(3)-83-40, Replacement of Yarway Reactor Water Level Switches, June 6, 1984; Modification M12-2(3)-83-36 through 39, Replacement of Pressure, Differential Pressure, and Flow Switches, Dresden Station Units 2 and 3, AIR 12-82-67.
2. USNRC IE Information Notice No. 83-08, Component Failures Caused by Elevated DC Control Voltage, 3/9/83.
3. Nutech Calculation XCE 004.0300.0128, Voltage Drop Calculation, 6/20/84.
4. Bechtel letter No.10802, Attachment 4, AGASTAT Relay Voltage, 1/16/86.

D5.3-1 (Deficiency) Motor Operated Valve Protection

DESCRIPTION: Motor operated valves are equipped with special motors designed to provide sufficient torque to operate the valve over its design range. These motors are typically short-time rated so that a smaller frame size can be used to reduce the weight. AC motors supplied on Limitorque motor operated valves on Dresden are 15 minute duty rated motors with a recommended locked rotor protection design time of 10 seconds. Dresden Station Procedure DMP-040-6 (Reference 1) requires all changes in MOV protection on safety-related valves to be approved by SNED. In such cases, Station Electrical Engineering Department (SEED) performs the actual design work.

The team reviewed the development of the overload heater selection by SEED for ac motor operated valve 3-1501-3A in support of Modification 85-58 (Reference 2) and had the following concerns:

1. SEED was using a handwritten draft procedure to establish recommended overload heater sizing that had no evidence of any reviews and/or approval (Reference 3), and which apparently was based upon a 1973 Limitorque recommendation (Reference 4). This procedure could not be used for selecting DC overloads because it did not include a DC calibration curve.
2. It appeared that only a few AC MOV overload relays were reviewed by SEED (only those requested by SNED) and that no DC MOV overload relay was ever reviewed by SEED. For relays not reviewed by SEED, there was no documented evidence of review by another organization to confirm the heater selection.
3. Neither the draft procedure nor the SEED recommendation to SNED mention any setpoint for the ±15% adjustment dial on the GE 124 K relays.
4. MOV 1501-3A locked rotor current supplied by Bechtel (Reference 5) and used by SEED did not agree with the station nameplate walkdown data sheets (Reference 6).
5. No backup heater sizing calculation existed in the SEED file, and the heater size included in the SEED recommendation could not be duplicated using the calculation procedure referenced in item 1.

BASIS: ANSI N45.2.11 (Reference 7) Section 4.2 requires that design analyses be performed in a controlled manner. Section 5.2 requires control of design interfaces such that changes to design information will be communicated between organizations. Section 6.3 requires that design reviews be performed to confirm that the inputs were correctly selected and an appropriate design method was used. Contrary to these requirements, design analyses were performed on safety-related components using draft procedures, and conflicting inputs, or not using all pertinent inputs.

REFERENCES

1. CECo Procedure DMP-040-6, Rev. 4, Safety-Related Motor Operated Valves, Data and Settings, 12/17/85.
2. CECo Modification M12-3-85-58, Replacement of Motor Operator MO-3-1501-3A.
3. CECo Procedure (No Number), Draft 2, Overload Relay Thermal Element Selection for Motor Operated Valves, Safety-Related.

4. Limitorque Dwg. 19-495-0006-3, Rev.0, Overload Relay Heater Sizing, 2/28/73.
5. Bechtel Letter G35-11-147, Motor Test Data for Heater Sizing, 9/19/83.
6. CECo Limitorque MOV Nameplate Data Checksheets 2-1501-3A, 2/20/85 and 3-1501-3A, 12/20/85.
7. ANSI N45.2.11, Quality Assurance Requirements for the Design of Nuclear Power Plants, 1974.
8. CECo Quality Procedure No. 3-1, Rev. 5, Design Control, 10/5/84.

D5.3-2 (Deficiency) Review of LPCI Room Cooler Motor Replacement

DESCRIPTION: Modifications M12-2(3)-84-49 and 50 (Reference 1) were prepared to replace non-qualified 5 hp General Electric fan motors with qualified Westinghouse motors for the LPCI Room Coolers. The new motors were purchased by Bechtel (Reference 2). Commonwealth Edison Company specification EM-20010 (Reference 3) requires that: 1) the motor deliver full load torque and horsepower without damage at 75% of rated voltage for one minute intervals, and 2) the motor bearings shall be sized for a 9 year "rating life" (L10) at rated speed.

The team reviewed the purchase requisitions, and did not find mention of the specification requirements in the purchasing documents, nor could Bechtel produce any evidence that these requirements could be met by the new motors.

BASIS: There was no evidence that the replacement motors complied with Specification EM-20010.

REFERENCES:

1. CECo Letter dated July 23, 1984, "Replacement of LPCI Room Cooler Motors 2(3)-5746A and B Modifications M12-2(3)-84-49 and 50, AIR-12-84-134."
2. Bechtel Power Corporation Material Requisition 13524-E-2E(Q) (FAN MOTORS) for Safety-Related Systems Equipment, Rev. 1, dated 7-19-82.
3. CECo System Standard EM-20010, "460 Volt Induction Motors," dated 8-1-75.
4. ANSI N45.2.11 Quality Assurance Requirements for the Design of Nuclear Power Plants, 1974.

D5.3-3 (Deficiency) Motor Operated Valve (MOV) Contactor Test Procedures

DESCRIPTION: Procedure DMP 7300-5 (Reference 1) was submitted to the NRC in response to SEP Topic III-10-A (Reference 2) to document the testing of MOV overload relays in safety-related circuits on a periodic basis. The procedure provides acceptance criteria and a test schedule. The team reviewed this test procedure and found:

1. The latest revision to this procedure (Reference 3) has broadened the acceptance criteria as follows:

   a. For standard trip CR123K heaters, the high acceptance limit was increased based upon GE curve GES7201A (Reference 4). This curve is different than the curve in GE Bulletin GEH2487C (Reference 5) used by CECo engineering to size overloads such that the field tested and accepted overload trip point may not provide the protection assumed in the engineering analysis.

   b. For quick trip CR123L heaters, the high acceptance limit was increased when the field test data gave results outside the range of that in GEH2487C. Again, the result is that the overload may not provide the required protection.

2. The test procedure is for 480 volt motor control centers, but the test schedule does not list all of the safety-related 480 V ac MOVs listed in DMP-040-6 (Reference 6). Those valves not listed include 1201-1A, 1201-4, and 2301-4.

3. No test procedure exists for 208 volt ac or dc MOVs (safety-related) listed in Procedure DMP-040-6.

BASIS: In the response to SEP Topic III-10-A, CECo committed to test all MOV overload relays in safety-related circuits on a periodic basis. As indicated above, there is no documented evidence that this commitment will be adequately met.

REFERENCES

1. CECo Procedure DMP7300-5, Rev. 1, Inspection & Maintenance of 480 V MCC Breakers and Contactors, 8/83.
2. NUREG-0823, Integrated Plant Safety Assessment Systematic Evaluation Program for Dresden Nuclear Power Station Unit 2, 2/83.
3. CECo Procedure DMP7300-5, Rev. 3, Inspection of 480 V MCC Breakers and Contactors, 9/84.
4. GE Curve GES7201A, Type CR124, Thermal Overload Relay Time Current Curve, 6/82.
5. GE Bulletin GEH2487C, CR124K Ambient Compensated Thermal Overload Relays, 8/79.
6. CECo Procedure DMP-040-6, Rev. 4, Safety-Related Motor Operated Valves Data and Settings, 12/17/85.
7. ANSI N45.2.11, Quality Assurance Requirements for the Design of Nuclear Power Plants, 1974.