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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

In the Matter of	)	
COMMONWEALTH EDISON COMPANY	)	Docket Nos. 50-237-SP
(Dresden Station, Units 2	)	50-249-SP
and 3)	)	(Spent Fuel Pool Modification)

Dear Administrative Judges,

Please find enclosed Commonwealth Edison Company's ("Commonwealth Edison") submittal responding to the NRC Staff's request for information regarding the resolution of certain Unresolved Safety Issues at the Dresden Nuclear Station, Unit 2. Although no exceptions have been filed, it is Commonwealth Edison's understanding that the Appeal Board is exercising its sua sponte authority to review the Licensing Board's final decision in this proceeding. Since the enclosed submittal is arguably relevant to Board Question No. 2, Commonwealth Edison is providing the submittal in accordance with the full disclosure requirements set forth in Duke Power Company (William B. McQuire Nuclear Station, Units 1 and 2), ALAB-143, 6 AEC 623 (1973). Commonwealth Edison does not believe the enclosed submittal draws into question the Licensing Board's resolution of Board Question No. 2. Nor should this letter indicate Commonwealth Edison's assent to the appropriateness of the Appeal Board's sua sponte review.

Respectfully submitted,

Robt G. Fitzgibbons Jr.
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RGF:kb

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September 3, 1982

Mr. Gus C. Lainas
Assistant Director for Safety
Assessment
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Dresden Station Unit 2
Unresolved Safety Issue
Status
NRC Docket No. 50-237

Reference (a): Gus C. Lainas letter to L. O. DelGeorge dated July 6, 1982.

Dear Mr. Lainas:

In Reference (a), Commonwealth Edison was requested to provide information regarding the resolution status of the following Unresolved Safety Issues at Dresden Unit 2:

(1) Waterhammer - (A-1)
(2) BWR Mark I Pressure Suppression Containments - (A-6, A-7, and A-39)
(3) Anticipated Transients Without Scram (A-9)
(4) BWR Nozzle Cracking - (A-10)
(5) Reactor Vessel Materials Toughness (A-11)
(6) Systems Interaction in Nuclear Power Plants (A-17)
(7) Environmental Qualification of Safety Related Electrical Equipment (A-24)
(8) Residual Heat Removal Requirements (A-31)
(9) Control of Heavy Loads Near Spent Fuel (A-36)
(10) Seismic Design Criteria (A-40)
(11) Pipe Cracks at Boiling Water Reactors (A-42)
(12) Containment Emergency Sump Reliability (A-43)
(13) Station Blackout (A-44)
(14) Shutdown Decay Heat Removal Requirements (A-45)
(15) Seismic Qualifications of Equipment in Operating Plants (A-46)
(16) Safety Implications of Control Systems (A-47)
(17) Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment (A-48)

Attachment 1 to this letter provides the requested information. In all cases we believe that continued operation is completely justified for the same reasons identified generically by the NRC. Where we have additional justification because of special plant features, procedures, reviews or modifications, we have described this in the attachment. We understand that this information will be used to support the full-term operating license conversion pending before the NRC.

To the best of my knowledge and belief the statements contained herein and in the attachment are true and correct. In some respects these statements are not based on my personal knowledge but upon information furnished by other Commonwealth Edison employees. Such information has been reviewed in accordance with Company practice and I believe it to be reliable.

Please address any questions you may have concerning this matter to this office.

One (1) signed original and thirty-nine (39) copies of this transmittal are provided for your use.

Very truly yours,

Thomas J Rausch

Thomas J. Rausch
Nuclear Licensing Administrator

lm

Region III Inspector - Dresden

SUBSCRIBED and SWORN to
before me this 3rd day
of September, 1982
Rosalie A. Printa
Notary Public

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Attachment I

TASK A-1 Waterhammer

1. Description of Problem

Waterhammer events are intense pressure pulses in fluid systems caused by any one of a number of mechanisms and system conditions such as rapid condensation of steam pockets, steam-driven slugs of water, pump startup with partially empty lines, and rapid valve motion. Since 1971 over 200 incidents involving waterhammer in pressurized and boiling water reactors have been reported. The waterhammers (or steam hammers) have involved steam generator feedrings and piping, the residual heat removal systems, emergency core cooling systems, containment spray, service water, feedwater and steam lines.

Most of the damage reported has been relatively minor, involving pipe hangers and restraints; however, several waterhammer incidents have resulted in piping and valve damage. The most serious waterhammer events have occurred in the steam generator feedrings of pressurized water reactors. In no case has any waterhammer incident resulted in the release of radioactive material.

Under Generic Task A-1, the potential for waterhammer in various systems is being evaluated and appropriate requirements and systematic review procedures are being developed to ensure that waterhammer is given appropriate consideration in all areas of licensing review. A technical report, NUREG-0582, "Waterhammer in Nuclear Power Plants" (July 1979), providing the results of an NRC staff review of waterhammer events in nuclear power plants and stating current staff licensing positions, completes a major subtask of Generic Task A-1.

2. Justification of Continued Operation

Although waterhammer can occur in any light water reactor and as approximately 118 actual and probable events have been reported in boiling water reactors as of September 1979, none have caused major pipe failures in boiling water reactors such as Dresden Unit 2 and none have resulted in the offsite release of radioactivity.

Dresden Unit 2 has installed several systems to preclude waterhammer from occurring in emergency core cooling system lines. The Low Pressure Coolant Injection (LPCI), High Pressure Coolant Injection (HPCI), and Core Spray (CS) systems all have a jockey pump to pressurize their respective lines and prevent the chance for any voids from forming. Also, a fill line routing water from the Diesel Generator Cooling Water System to the Containment Cooling Service Water (CCSW) subsystems of LPCI has been installed to prevent voids from forming in these lines. In addition to these waterhammer precautions, a drain line has been installed on the lowest elevation of the HPCI turbine inlet main steam line to remove possible condensate and preclude waterhammer damage from occurring.

In the event that Task A-1 identifies potentially significant waterhammer scenarios which have not explicitly been accounted for in the design and operation of Dresden, Unit 2 corrective measures will be considered at that time. This task has not identified the need for measures beyond those already implemented.

Based on the foregoing, we conclude that Dresden Unit 2 can be operated prior to ultimate resolution of this generic issue without undue risk to the health and safety of the public.

TASK A-6 Mark 1 Short Term Program
TASK A-7 Mark 1 Long Term Program

1. Description of Problem

During the conduct of a large scale testing program for an advanced design pressure-suppression containment system (Mark III) for BWRs, new suppression pool hydrodynamic loads associated with a postulated loss-of-coolant accident (LOCA) were identified which had not been explicitly included in the original design of the Mark I containment systems. These additional loads result from dynamic effects of drywell air and steam being rapidly forced into the suppression pool (torus) during a postulated LOCA event. In addition, recent experience at operating plants has indicated that the dynamic effects of safety-relief valve (SRV) discharges to the suppression pool could be substantial and should be reconsidered.

The results of the Mark I containment short-term program (STP) have provided assurance that the Mark I containment system of each operating BWR facility would maintain its integrity and functional capability during a postulated LOCA. However, the STP evaluation was conducted using a "most probable load" approach which was aimed at the identification of load magnitudes and load combinations which were most likely to be encountered during the course of a postulated design basis LOCA. In addition, the STP structural acceptance criteria were selected to assure that, for the most probable loads induced by a postulated design basis LOCA, a safety factor to failure of at least two existed for the weakest structural or mechanical component in the containment system for each operating Mark I BWR facility.

Consequently, since the design margin of safety for the containment systems of operating Mark I facilities has been reduced from the margin believed to be present at the time these facilities were originally reviewed and licensed, the need existed (1) to establish design basis LOCA loads which are appropriate for the life of the facility, and (2) to restore the originally-intended design safety margins for the containment systems.

2. Justification for Continued Operation

The safety issue addressed by this Task Action Plan (TAP) is applicable to boiling water reactor (BWR) facilities with the Mark I containment system design. A total of 25 such facilities have been built or are being built in the United States; of these, 22 are currently licensed for operation.

For Mark I BWRs currently licensed for operation, the NRC has concluded that there is reasonable assurance that continued operation, pending completion of this task, does not constitute an undue risk to the health and safety of the public for the following reasons:

As documented in NUREG-0408, "Mark I Containment Short-Term Program Safety Evaluation Report," December 1977, based upon our review of the generic "Short-Term Program Final Report" and Addenda submitted by the Mark I Owner's Group and the plant-unique analysis reports submitted by each licensee of an operating Mark I BWR facility, we have concluded that licensed Mark I BWR facilities can continue to operate safely, without undue risk to the health and safety of the public, during an interim period of approximately 2 years, while a methodical, comprehensive Long-Term Program (LTP) is conducted. This conclusion has been made based on NRC determinations (1) that the magnitude and character of each of the hydrodynamic loads resulting from a postulated LOCA have been adequately defined for use in the Short Term Program (STP) structural assessment of the Mark I containment system, (2) that, for the "most probable" loads induced by a postulated LOCA, a safety factor to failure of at least two exists for the weakest structural or mechanical component in the containment system for each operating Mark I BWR facility, and (3) that, based on (1) and (2), each Mark I containment system would maintain its integrity and functional capability in the unlikely event of a design basis LOCA.

The NRC has reviewed the Mark I Owner's Program Action Plan for the LTP and have found that it is reasonably designed to provide resolution of the issues raised during our review of the STP and to satisfy the LTP objectives. The NRC is continually monitoring the progress of the LTP to assure that these requirements are satisfied.

As was the case during the conduct of the STP, if information becomes available during the course of the LTP which indicates that the safety factor to failure of a component of the containment system of a Mark I BWR facility is less than two, immediate corrective action could be required. Such action could take the form of structural modification, installation of load mitigating devices, or other appropriate measures.

3. Program to Resolve Issue

As previously stated the NRC has reviewed the "Mark 1 Owners Group Long-Term Program". They issued their assessment of the program as NUREG -0651 "Mark 1 Containment Long-Term Program Safety Evaluation Report, Resolution of Generic Technical Activity A-7", dated July, 1980. Commonwealth Edison Company is addressing the issues as presented in NUREG - 0661 on a schedule commensurate with order dates specified by the Commission in a letter from Dominic Vassalo to Mr. DelGeorge dated January 19, 1982 and as clarified in a letter from T.J. Rausch to Mr. Denton dated April 6, 1982.

TASK A-39 SRV, Pool Dynamic Loads

1. Description of Problem

BWR plants are equipped with relief valves that discharge into the wetwell. Upon relief valve actuation, the initial air column within the SRV discharge line is accelerated by the high pressure steam flow and expands as it is released into the pool as a high pressure air bubble. The high rate of air and steam injection flow in the pool followed by expansion and contraction of the bubble as it rises to the pool surface produces pressure oscillations on the pool boundary. This effect is referred to as the air-clearing phenomenon.

Experience at several BWR plants with pressure suppression containments has shown that damage to certain wetwell internal structures can occur during safety/relief valve (SRV) blowdowns as a result of air-clearing and steam quenching vibration phenonmena.

In addition to the boundary loads, e.g., containment structures, reactor pedestal, the air injection and subsequent bubble motion produces pressure waves and water movement within the pool that produce drag loads on components in the pool.

Following the air-clearing phase, pure steam is injected into the pool. Condensation oscillations occur during this time period. However, the amplitudes of these vibrations are relatively small at low pool temperatures. Continued blowdown into the pool will increase the pool temperature until a threshold temperature is reached. At this point, steam condensation becomes unstable. Vibrations and forces can increase by a factor of 10 or more if the SRV continues to blow down. This effect is referred to as the steam quenching vibration phenomenon. Current practice for the BWR operating plants is to restrict the allowable operating temperature envelope via Technical Specifications such that the threshold temperature is not reached.

In response to the concern on relief valve loads, letters were sent in 1975 to all licensees of operating BWR plants requesting that they report on the potential magnitude of relief valve loads, and on the structural capability of the suppression chamber and internal structures to tolerate such loads. As a result of the generic concerns, owners groups were formed by both Mark I and II utilities. Through these groups, integrated generic analytical and experimental programs have been developed to address the subject of SRV loads.

With respect to Mark III containments, the staff has established acceptance criteria for quencher loads. These criterion were conservatively established based on the data base available to us.

One of these criteria requires the applicants to assume that, for the events involving multiple valve actuations, the bubbles from each SRV discharge reach their peak pressures simultaneously and then oscillate in phase. GE, however, believes that this assumption is unrealistically conservative and will impose unduly severe loadings on equipment and piping in the plant. In early 1978, GE proposed an equipment reevaluation program, which considers a statistical approach to determine the effects of bubble phasing considerations. This Containment Loads Report - Mark III Containment," Rev. 2. Since this approach is expected to be applied generically to all plants with Mark III containments, we have included this review item in the task action plan.

Recently, GE issued a Part 21 notification related to consecutive actuation of multiple safety/relief valves and concomitant load increases for BWR Mark III water pressure-suppression containments. This concern resulted from a recent study performed by GE of the primary system pressure response following an isolation event. The results showed that more than one safety/relief valve could be actuated consecutively, as a result of a reactor isolation event. This SRV load combination has not been considered in the design. Discussions with GE have also revealed that this concern is generic to all BWR containments and, therefore, is included in the task action plan.

Results of this task will be incorporated into Tasks 1.c and 2.d for establishing SRV load cases and load combination. A report of our evaluation will also be issued for this particular concern.

2. Justification for Continued Operation

As discussed in Section 1, the safety issue addressed by this task is the possible damage to wetwell internal structures and the pool boundary that could occur due to air-clearing and steam quenching phenomena resulting from safety relief valve (SRV) discharge into the suppression pools of BWR plants. It is of concern to all BWR plants using the Mark I, Mark II, or Mark III pressure suppression type containments.

This task will provide the basis for establishing acceptance criteria for safety relief valve loads and for suppression pool temperature limits. In conjunction with Task A-7 (Mark I Long-Term Program) and Task A-8 (Mark II Containment Pool Dynamic Loads), a complete evaluation will be provided of suppression pool dynamic loads for BWR containments.

For Dresden 2 the justification for continued operation and licensing is based on an evaluation of operating experiences and the plant capability to tolerate SRV loads in the short term. SRV operating experience has shown that in all but a few instances, SRV discharges have performed satisfactorily without any evidence of

damage either due to the hydrodynamic loads or pool temperature effect. In these cases where localized damage occurred at other plants, the damage did not result in a loss of the containment function, or release of radioactivity, or undue risk to the health and safety of the public. In those cases, repairs were made and additional margin was included in the structures. With respect to the plant capability, the NRC has concluded that the plants have the capability to tolerate SRV loads because the loads are related to the structural fatigue life. However, all plants will be required to demonstrate the capability to meet the SRV loads criteria and pool temperature limit which will be established by this task.

In summary, it has been concluded that the SRV loads are related to the structural fatigue life. Therefore, CECo feels that the plants with Mark I containment can be allowed to continue operation until completion of the Mark I Long-Term Program.

3. Program to Resolve Issue

The NRC assessment of the Mark I Long-Term Program (NUREG-0661) addresses the SRV discharge loads and specifies requirements for torus temperature monitoring. Therefore, these issues will be resolved on a schedule as reported in Section 3 of Tasks A-6 and A-7.

TASK A-9 Anticipated Transients Without Scram

1. Description of Problem

Nuclear plants have safety and control systems to limit the consequences of temporary abnormal operating conditions or "anticipated transients." Some deviations from normal operating conditions may be minor; others, occurring less frequently, may impose significant demands on plant equipment. In some anticipated transients, rapidly shutting down the nuclear reaction (initiating a "scram"), and thus rapidly reducing the generation of heat in the reactor core, is an important safety measure. If there were a potentially severe "anticipated transient" and the reactor shutdown system did not "scram" as desired then an "anticipated transient without scram," or ATWS would have occurred.

All boiling water reactors, including Dresden Unit 2 have been required to provide recirculation pump trip in the event of a reactor trip and to provide additional operator training for recovery from anticipated transient without scram events.

2. Justification for Continued Operation

A Recirculation Pump Trip (RPT) provision has been incorporated into the Dresden Unit 2 design. An Alternate Rod Injection (ARI) subsystem is currently being installed and is scheduled for completion by April, 1983. A commitment has also been made to modify the scram discharge system to include two instrument volumes that will incorporate diverse and redundant instrumentation. This work is scheduled for completion by December, 1984. Emergency procedures and operator training to cope with potential anticipated transient without scram events have been implemented. These procedures and training will be revised as the ARI subsystem and scram discharge system modifications are completed. Operator training and action as described in the Cordell Reed letter to H. R. Denton dated March 16, 1982, significantly improved the capability of the facility to withstand a range of anticipated transient without scram events.

The anticipated transient without scram rulemaking is currently scheduled for completion by Fall 1982. Based on our review, we feel that there is reasonable assurance that Dresden Unit 2 can be operated prior to ulimate resolution of this generic issue without endangering the health and safety of the public.

TASK A-10 BWR Nozzle Cracking

I. Description of Problem

A. BWR Feedwater Nozzle Cracking

Of the 23 operating BWRs with feedwater nozzle/sparger systems (normally 4 nozzles/spargers per BWR, nominal nozzle diameter being (10"-12")), 21 have been inspected to date (1-25-79) resulting in the discovery of blend radius or bore cracking in all but three vessels. Although most cracks have been in the range of 1/2" to 3/4" total depth (including cladding), one crack penetrated the cladding into the base metal for a total depth of approximately 1.50 inches. The initiation of cracking is due to high cycle fatigue caused by fluctuations in water temperature within the vessel in the sparger-nozzle region during periods of low feedwater temperature when the flow may be unsteady and intermittent. Once initiated, the cracks are driven deeper by the larger pressure and thermal cycles associated with startup and shutdown.

Fracture analyses indicate that the cracks found to date in the feedwater nozzles constitutes a potential safety problem because the observed rate of crack growth with time in service is such that the margin of safety against fracture will be reduced below aceptable values unles the cracks are detected and ground out every few years. Obviously, repair by grindout can be repeated only a few times before ASME Code limits for nozzle reinforcement are exceeded. However, repair by welding buildup of the grindout has not been demonstrated to be acceptable. In addition, the inspection and removal of cracks by grinding has caused enough radiation exposure to personnel to be deemed unacceptable as a long-term solution.

B. Control Rod Drive Hydraulic Return Line Nozzle Cracking (CRDRL Nozzle)

Each of the 22 applicable BWRs has one CRDRL nozzle of 3"-4" diameter, which is normally located approximately 4 feet below the level of the feedwater nozzles (in the Oyster Creek and Nine Mile Point vessels, the CRDRL nozzle is located at the same level as the feedwater nozzles). Thermal fatique cracks have been found by dye penetrant (PT) inspection of the CRDRL nozzle and the area immediately beneath the nozzle at 12 units inspected to date (1-25-79). These cracks resemble those found in the BWR feedwater nozzles, and the cause of cracking appears to be thermal fatigue. All but two of the operating domestic BWRs have some sort of thermal sleeve (there are several designs) in the CRDRL nozzle, but because of the limited number of inspections of nozzles with sleeves, the efficiency of the sleeves is not known.

To date, the principal activity of licensees has been to reroute or temporarily valve out the CRDRL. Although both accomplish the intended purpose of shutting off cold water flow to the nozzle, General Electric Company (GE) has further recommended that the CRD system be operated in an isolated mode. GE recommends against retention of the present CRDRL, even valved out, because of the potential for stress corrosion in the stagnant line. GE also recommends against operation with a rerouted CRDRL open to the reactor vessel. The recommendation to isolate the rerouted line was made on the basis that return to the vessel is unneccessary for proper CRD system operation and that CRD makeup capability to the vessel will be maintained even when the return line is eliminated entirely.

The staff still considers the matter of CRDRL isolation to be an unresolved issue because of questions regarding the amount of CRD pump flow which will be available to the vessel, the possible effects of isolation upon various drive parameters, and recently reported potential long-term deleterious effects on certain components of the CRD hydraulic system. GE has begun an evaluation of component performance of affected portions of the CRD hydraulic system and has commenced investigation of possible system modifications. The staff must assess these proposals prior to completion of its review of this subject. In the interim, the staff will review control rod test information from each facility which has modified its present CRD system by valving out or rerouting. Additionally, to increase assurance of safety for continued operation, that staff is recommending inspections of the CRDRL nozzle blend radius and bore at each BWR during its next scheduled refueling outage. As in the case of feedwater nozzles, we are especially concerned, particularly in the case of older units, that a potential safety problem could raise from deep cracks which would necessitate weld repair.

2. Justification for Continued Operation

The staff anticipates that this task will result in long-term solutions that will provide: (1) assurance that a conservative margin of safety against vessel failure due to nozzle cracks in maintained at operating facilities, (2) more stringent licensing requirements concerning selection of materials and design for nozzles, thermal sleeves, and spargers; (3) more stringent inservice inspection and repair criteria; (4) modification of physical systems and/or operating procedures to minimize the occurrence of crack initiation and propagation; and (5) reliable inservice inspection techniques for detection of nozzle flaws from positions exterior to the reactor vessel.

With respect to feedwater nozzle cracking, specific long-term corrective measures will include system and operational changes to reduce the feedwater to reactor water temperature differential during low power operation, an improved thermal sleeve-sparger design to reduce bypass flow which exposes the nozzle surface in fluctuating water temperatures, and removal of clad from the nozzle surface which is believed to provide a surface more resistant to fatigue cracking. Implementing some combination of these measures after plants are already under construction or are operating is feasible, e.g., several utilities with operating reactors have already implemented clad removal and the first new thermal sleeve-sparger design has been installed in an operating plant.

With respect to control rod drive return line nozzle cracking, specific long-term corrective measures will include system modifications that assure proper control rod drive system performance with the return line isolated (if one is installed by design) or eliminated by design. Control rod drive return line isolation has been implemented at several operating faciliities as an interim corrective measure. Studies are currently underway to determine the acceptability of long-term operation in this manner. If these studies (which are scheduled for completion in early 1979) demonstrate no degradation of affected components, no further action in this regard will be necessary for plants so modified.

During the time period required to develop the long-term solutions under this task, interim measures have been taken. Specifically, as required by the NRC inservice inspection using liquid penetrant examinations are being performed in accordance with the procedures and acceptance criteria set forth in detail in NUREG-0312, "Interim Technical Report on BWR Feedwater and Control Rod Drive Return Line Nozzle Cracking," July 1977. Edison is also utilizing ultrasonic inspection techniques in an effort to develop effective techniques that will allow early detection of subsurface flaws. Enhancement of ultrasonic testing techniques will substantially reduce personnel exposures. The scheduling and extent of inspection is based upon conservative estimates of crack growth from fracture mechanics analyses assuming undetected flaws. Scheduling is thus dependent upon the reactor's record of past repair (grindouts, clad removal, etc.), operating history (number of startup/shutdown cycles since dye-penetrant inspection), and licensee actions to minimize crack initiation by procedural or mechanical change.

The staff has been actively involved in reviewing and approving the results of nozzle inspections and remedial actions proposed by licensees to assure continued safe operation. To date the extent of nozzle cracking at operating plants has been limited to depths which can be removed by grinding without exceeding ASME Code limits for nozzle reinforcement.

In addition the staff has suggested that measures be taken at affected operating plants and by applicants for plants in the operating license review stage prior to opeation, to minimize the occurrence of conditions conductive to crack initiation and growth. These measures include monitoring feedwater temperatures and flow, minimizing rapid changes in feedwater flow and temperature, minimizing the duration of cold feedwater injection, avoiding inadvertent or unnecessary HPCI injection, avoiding the unnecessary introduction of cold water from the reactor water cleanup system, and eliminateing flow through the control rod drive return line (after assuring proper system operation in an isolated mode). Although cracking of the pressure vessel nozzles is important to safety, NRC staff analyses indicate that cracking that has penetrated the vessel cladding will grow at a slow enough rate such that the cracking does not pose a critical safety concern today that warrants immediate action. Rather, the staff believes that sufficient time is available, due to the conservative design of the reactor pressure vessel, to permit continued operation of the affected facilities while studies on these events continued on schedule.

Based on the interim measure being taken at operating facilities and the design margins available in the reactor pressure vessel, we have concluded that operation of such facilities does not present an undue risk to the health and safety of the public.

3. Program to Resolve Issue

NUREG-0619 "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking" contains the NRC required actions to resolve the subject safety issue. The following are references which define the Commonwealth Edison program schedule and action for addressing the NUREG requirements.

1. Letter from R.F. Janecek to Mr. Eisenhut dated February 23, 1981.
2. Letter from T.J. Rausch to Mr. Eisenhut dated November 6, 2981.
3. Letter from T.J. Rausch to Mr. Eisenhut dated February 23, 1982.

TASK A-11 Reactor Vessel Material Toughness

1. Description of Problem

Resistance to brittle fracture is described quantitatively by material property generally denoted as "fracture toughness." Fracture toughness has different values and characteristics depending upon the material being considered. For steels used in a nuclear reactor pressure vessel, three considerations are important. First, fracture toughness increases with increasing temperature; second, fracture toughness decreases with increasing load rates; and third, fracture toughness decreases with neutron irradiation.

In recognition of these considerations, power reactors are operated within restrictions imposed by the Technical Specifications on the pressure during heatup and cooldown operations. These restrictions assure that the reactor vessel will not be subjected to a combination of pressure and temperature that could cause brittle fracture of the vessel if there were significant flaws in the vessel material. The effect of neutron radiation on the fracture toughness of the vessel material over the life of the plant is accounted for in Technical Specification limitations.

The principal objective of Task A-11 is to develop safety criteria to allow a more precise assessment of safety margins during normal operation, transients and accident conditions in older reactor vessels with marginal fracture toughness.

2. Justification for Continued Operation

CECo's letter dated March 31, 1982, from T.J. Rausch to R.R. Denton transmits proposed Tech. Spec. changes for Dresden Unit 2 regarding reactor vessel toughness.

In the letter it is emphasized tha the beginning inherent jet pump water gap results in low end of life fluences and subsequently insignificant shift in the transition temp. due to irradiation.

Furthermore it is estimated that the fluence level of $2X10^{17}$ n/cm (E>, 1 mev.) in the Reg. Guide 1.99 graph for implementation of Appendix G, pressure temp. requirements will not be reached until 10 Effective Full Power Years. In addition the first Appendix H capsule date will soon become available providing actual shifts in RTNDT.

Based on the above the RPV has adequate toughness for achieved experienced under the current specification.

Therefore, based upon the foregoing, we conclude that Dresden can be operated prior to resolution of this generic issue without undue risk to the health and safety of the public.

TASK A-17 Systems Interaction in Nuclear Power Plants

1. Description of Problems

In November 1974, the Advisory Committee on Reactor Safeguards requested that the NRC staff give attention to the evaluation of safety systems from a multi-disciplinary point of view, in order to identify potentially undesirable interactions between plant systems. The concern arises because the design and analysis of systems is frequently assigned to teams with functional engineering specialities -- such as civil, electrical, mechanical, or nuclear. The question is whether the work of these functional specialists is sufficiently integrated in their design and analysis activities to enable them to identify adverse interactions between and among systems. Such adverse events might occur, for example, because designers did not assure that redundancy and independence of safety systems were provided under all conditions of operation required, which might happen if the functional teams were not adequately coordinated. Task A-17 was initiated to confirm that present review procedures and safety criteria provide an acceptable level of redundancy and independence for systems required for safety by evaluating the potential for undesirable interactions between and among systems.

2. Justification for Continued Operation

Current CECo review procedures and safety criteria provide reasonable assurance that an acceptable level of redundancy and independence is provided for systems that are required for safety. Furthermore, approximately 40 years of reactor operating experience at Dresden and Quad Cities Stations have shown no adverse systems interaction exist.

Therefore, CECo concludes that there is reasonable assurance that Dresden Unit 2 can be operated prior to the final resolution of this generic issue without endangering the health and safety of the public.

<u>Task A-24 Environmental Qualification of Safety Related Electrical Equipment</u>

1. Description of Problem

The NRC through I.E. Circular 78-08, Bulletin 79-01, Bulletin 79-01B and Supplements, and Commission Memorandum and Order (CHI-80-21) brought to the attention of the nuclear industry a potential problem - the environmental qualification of Class IE Equipment. These NRC actions require the review of Class IE Equipment subjected to a harsh environment (resulting from a LOCA or HELB) to determine if the equipment can fulfill its design function in such an accident environment.

2. Justification for Continued Operation

CECo has provided the NRC with four major submittals which address justification for continued operation. The original System Component Evaluation Work sheets, reference 1 below, and most recently the SCEW sheets associated with TMI Section Plan Equipment, reference 2, contain such justification for those pieces of equipment lacking proper documentation. In response to the NRC request for additional justification resulting from preliminary reviews of CECo submittals, reference 3 was submitted supporting continued Dresden Unit 2 operation. Finally, in response to the NRC's SER and Frankline's TER CECo submitted additional material justifying continued operation of Dresden 2 until equipment testing and/or replacement programs are completed.

1. "Dresden Station Unit 2 Response to NRC Order Concerning Environmental Qualification of Class IE Electrical Equipment," J.S. Abel to D.G. Eisenhut, October 31, 1980, NRC Docket No. 50-237.

2. "Dresden Station Units 2&3 and Quad Cities Station Units 1&2 Environmental Qualification of TMI Action Plan Equipment," E.D. Swartz to D.G. Eisenhut, August 2, 1982, NRC Docket Nos. 50-237/249 and 50-254/265.

3. "Dresden Station Unit 2, Environmental Qualification of Safety Related Electrical Equipment," J.S. Abel to H.R. Denton, dated March 2, 1981, NRC Docket No. 50-237.

4. "Dresden Stations Units 2&3 and Quad Cities Stations Units 1&2 Environmental Qualification of Safety Related Class IE Electrical Equipment," T.J. Rausch to H.R. Denton, dated September 4, 1981, NRC Docket Nos. 50-237/249 and 50-254/265.

3. Corrective Action

For equipment lacking proper qualification documentation a schedule addressing equipment testing and/or replacement has been developed. This schedule is commensurate with the NRC draft schedule for completion of equipment qualification (the end of the schedule refueling outage originating after March 31, 1982). The test program is presently under way at Wyle Laboratories and is targeted for completion by Mid-1983. Owner's Group testing of presently installed equipment is about to commence and is targeted for completion by the Fall of 1983. Equipment replacement and relocation efforts are underway and will result in equipment changeouts begining during the presently scheduled refueling outage commencing in January, 1983. The existing surveillance and maintenance program is being reviewed to insure compliance with qualification requirements.

TASK A-31 Residual Heat Removal Requirements

1. Description of Problem

Long term cooling of the reactor coolant system is required in order to perform inspection and repairs. For this reason, the ability to transfer heat from the reactor to the environment after shutdown is an important safety function. It is essential that a power plant have the capability to remain in the cold-shutdown condition on a long term basis.

2. Justification for Continued Operation

The shutdown cooling system is available after the reactor coolant system has been sufficiently cooled and depressurized. The design of the system is based on cooling the reactor coolant system from 350°F to 125°F within 24 hours after shutdown. The system consists of three loops which tie into the suction line of the reactor recirculation pumps. Each loop contains one pump and one heat exchanger which is cooled by the reactor building closed cooling water. Operating experience has shown that, at only eight hours after normal (main condenser) shutdown, only one pump and one heat exchanger are necessary to cool down. The flow path continues into the low pressure coolant injection system and then into each of the reactor recirculation loops. Various temperature and pressure interlocks must be met before startup of the system. There is diversity in the AC, DC, and emergency diesel power supplies to assure system isolation and protection.

The low pressure coolant injection system can be used if the shutdown cooling system is inoperable. The low pressure coolant injection system is capable of injecting cooling water from the suppression pool or the contaminated condensate storage tank. This water is cooled by passing through the heat exchangers which are cooled by the containment cooling service water. The other modes of the low pressure coolant injection system are suppression pool cooling, containment spray and suppression pool spray. The suppression pool cooling and suppression pool spray will condense steam and maintain the water temperature during regular operation and accident conditions. The containment spray will cool the air space inside primary containment through a spray ring header.

The low pressure coolant injection system and containment cooling service water system have redundant trains. The low pressure coolant injection pumps are powered from essential service buses, and all motor-operated valves are powered from essential service motor control centers and are also accessible for manual operation if needed.

Based on the above, we conclude that Dresden Unit 2 can be operated prior to ultimate resolution of this generic issue without undue risk to the health and safety of the public.

TASK A-36 Control of Heavy Loads Near Spent Fuel

1. Description of Problem

In nuclear plant operation, maintenance and refueling activities, heavy loads may be handled in several plant areas. If these loads were to drop, they could impact on stored spent fuel, fuel in the core, or equipment that may be required to achieve safe shutdown or permit continued decay heat removal. If sufficient stored spent fuel or fuel in the core were damaged and if the fuel is highly radioactive due to its irradiation history, the potential releases of radioactive material could result in offsite doses that exceed 10CFR Part 100 limits. If the load damaged equipment associated with redundant or dual safe shutdown paths, the capability to achieve safe shutdown may be defected.

In this task a heavy load is defined as a load whose weight is greater than the combined weight of a single spent fuel assembly and its handling tool.

Task A-36 was established to systematically examine staff licensing criteria and the adequacy of measures in effect at operating plants, and to recommend necessary changes to assure the safe handling of heavy loads once a plant becomes operational.

Additionally, with the increased spent fuel storage capacities at many operating plants, largely in the form of increased density of fuel storage within the pool, the potential for a given load to damage a large number of fuel assemblies has increased.

2. Justification for Continued Operation

As a result of NUREG-0612, CECo. has made several submittals on control of heavy loads at Dresden. On February 22, 1982, a telephone conference call was held between the NRC Staff, the Franklin Research Center (FRC) and CECo to discuss FRC draft Technical Evaluation Reports (TER's) concerning "Phase I" control of heavy loads at our Dresden and Quad Cities Stations.

As a result of our review of the draft TER's, and to document the conference call discussions, CECo provided their response to each concern and recommendation that was identified by the FRC in their TER's. From the response, along with the initial submittals for Dresden and Quad Cities, it is our understanding that this will form the basis for a final Technical Evaluation Report from FRC for each station and ultimately the NRC Staff Safety Evaluation of "Phase I" of this issue.

Therefore CECo concludes that there is reasonable assurance that Dresden Unit 2 can be operated prior to the final resolution of this generic issue without endangering the health and safety of the public.

TASK A-40 Seismic Design Criteria - Short Term Program

1. Description of Problem

The seismic design process required by current NRC criteria includes the following sequence of events.

A. Define the magnitude or intensity of the earthquake which will produce the maximum vibratory ground motion at the site (the safe shutdown earthquake or SSE).

B. Determine the free-field ground motion at the site that would result if the SSE occurred.

C. Determine the motion of site structures by modifying the free-field motion to account for the interaction of the site structures with the underlying foundation soil.

D. Determine the motion of the plant equipment supported by the site structures.

E. Compare the seismic loads, in appropriate combination with other loads, on structures, systems, and components to safety, with the allowable loads.

While this seismic design sequence includes many conservative factors, certain aspects of the sequence may not be conservative for all plant sites. At present it is believed that the overall sequence is adequately conservative. The objective of this program is to investigate selected areas of the seismic design sequence to determine their conservatism for all types of sites, to investigate alternate approaches to parts of the design sequence, to quantify the overall conservatism of the design sequence, and to modify the NRC criteria in the Standard Review Plan if changes are found to be justified. In this manner this program will provide additional assurance that the health and safety of the public is protected, and if possible, reduce costly design conservatisms by improving (1) current seismic design requirements, (2) NRR's capability to evaluate the adequacy of seismic design of operating reactors and plants under construction, and (3) NRR's capability to quantitatively assess the overall adequacy of seismic design for nuclear plants in general.

2. Justification For Continued Operation

The objective of the aforementioned task is to investigate selected areas of seismic design to determine the conservatism for Dresden, to investigate alternate approaches to parts of the design sequence, to qauntify the overall conservatism of the design sequences and to modify the licensing criteria if changes are found to be justified. The results of the task will be applicable to Dresden Nuclear Power Station.

It is anticipated that the results of this task will provide confirmation that current requirements provide an overall conservative approach to seismic design. The general result that is anticipated from this task is the development of better insight into seismic design considerations that will permit establishment of a set of integrated requirements providing for more realistic and effective designs without a loss of overall margin.

Three general types of results are expected from this task. The first is the ability to select seismic design ground motion inputs for Dresden that are more appropriate for the site and thus will result in a more consistent level of seismic design.

Second, it is expected that these investigations will demonstrate that the current methods of analysis are conservative in relation to other methods that could be justified and to provide a quantitative idea of how conservative they are. Third, it is expected that this effort will demonstrate that the overall safety margins attained using current methods are considerable. In the interim, it is believed that continuation of the current licensing requirements will assure an acceptable level of safety in plant seismic design.

If the results of this task action plan are not as anticipated and the current criteria prove not to be adequately conservative, these results will not affect the seismic design criteria at Dresden; because the original design criteria as specified in the FSAR are generally more conservative than the current design criteria.

Based on the discussion above, it is concluded that while this task is being performed, continued operation and plant licensing can proceed with reasonable assurance of protection to the health and safety of the public.

3. Added Technical Justification

The Systematic Evaluation Program (SEP) is currently in progress at Dresden Station. The last SER written June 30, 1982, Docket No. 50-237, LS05-82-06-130, SEP SAFETY TOPIC No. III-6, TITLE Seismic Design Considerations, identified three concerns as identified in the following.

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

1. Piping Systems - The staff has identified deficiencies regarding the existing piping supports. Therefore, we are unable to conclude that safety-related piping systems are capable of withstanding the postulated SSE loads.

2. Mechanical Equipment - The staff lacked sufficient design information for 3 out of 9 of the mechanical equipment items sampled. Therefore, the staff cannot conclude that mechanical equipment is adequately designed.

3. Programs undertaken by the SEP Owners Group are intended to provide a set of general analytical methodologies for the seismic qualification of cable trays and for documentation of the functionability of other safety-related electrical equipment subjected to seismic loads; these programs have not been completed.

Item 1 is being satisified by I&E review of Bulletin 79-14. As a result, piping supports are being modified and added to safety-related piping systems to upgrade the systems to withstand postulated SSE loads.

The current schedule to complete all modification work is December 31, 1983. For the interim, operability studies have been completed on safety-related piping and have demonstrated that Dresden can be safely shutdown during a postulated SSE event.

Item 2 Sargent and Lundy completed a special study on the subject mechanical equipment, which demonstrated that the subject mechanical equipment was adequately designed.

Item 3 is still in progress URS/John A. Blume and Associates are performing analyses to qualify cable trays and developing documentation to demonstrate the functionability of other safety-related electrical equipment subject to seismic loads. This study is scheduled to be completed in October 1982.

Based on the SEP conclusions generated thus far and results of IE Bulletin 79-14, continued operation and plant licensing can proceed with reasonable assurance of protection to the health and safety of the public.

TASK A-42 Pipe Cracks at Boiling Water Reactors

1. Description of Problem

Pipe cracking has occurred in the heat affected zones of welds in primary system piping in boiling water reactors (BWRs) since the mid-1960's. These cracks have occurred mainly in Type 304 stainless steel, which is the type used in most operating BWRs. The major problem is recognized to be intergranular stress corrosion cracking (IGSCC) of austenitic stainless steel components that have been made susceptible to this failure by being "sensitized," either by post-weld heat treatment or by sensitization of a narrow heat affected zone near welds.

"Safe ends" (short transition pieces between vessel nozzles and the piping) that have been highly sensitized by furnace heat treatment while attached to vessels during fabrication were very early (late 1950's) found to be susceptible to IGSCC. Because of this, the Atomic Energy Commission took the position in 1969 that furnace-sensitized safe ends should not be used on new applications. Most of the furnace-sensitized safe ends in older plants have been removed or clad with a protective material, and there are only a few BWRs that still have furnace-sensitized safe ends in use. Most of these, however, are in smaller diameter lines.

Earlier reported cracks (prior to 1975) occurred primarily in 4-inch diameter recirculation loop-bypass lines and in 10-inch diameter core spray lines. More recently cracks were discovered in recirculation riser piping (12-inch to 14-inch) in foreign plants. Cracking is most often detected during Inservice Inspection using ultrasonic testing techniques. Some piping cracks have been discovered as a result of primary coolant leaks.

In response to these occurrences of BWR primary system cracking, the NRC has taken a number of measures. These actions included:

Issuance of Regulatory Guide 1.44 on "Control of the Use of Sensitized Stainless Steel."

Issuance of Regulatory Guide 1.45 on "Reactor Coolant Boundary Leak Detection Systems."

Closely following the incidence of cracking in BWRs, including foreign experience.

Encouraging replacement of furnace-sensitized safe ends.

Requiring augmented in-serivce inspection (additional more frequent ultrasonic examination) of "service sensitive" lines, i.e., those that have experienced cracking.

Requiring upgrading of leak detection systems.

2. Justification for Continued Operation

CECo's letter from L.O. DelGeorge to D.G. Eisenhut, dated 7/7/81, and its references addresses this problem as developed by NUREG-0313, Rev. 1. CECo feels, because of the leak-before-break criterion, that pipe cracking is an availability problem and is not a safety issue.

Therefore, based upon the foregoing, we conclude that Dresden can be operated prior to resolution of this generic issue without undue risk to the health and safety of the public.

TASK A-43 Containment Emergency Sump Reliability

1. Description of Problem

Following a postulated loss-of-coolant accident, i.e., a break in the reactor coolant system piping, the water flowing from the break would be collected in the suppression pool. This water would be recirculated through the reactor system by the emergency core cooling pumps to maintain core cooling. This water may also be circulated through the containment spray system to remove heat and to draw water from the suppression pool could disable the emergency cooling and containment spray systems.

The concern addressed by this Task Action Plan for boiling water reactors is limited to the potential for degraded emergency core cooling system performance as a result of thermal insulation debris that may be blown into the suppression pool during a loss-of-coolant accident and cause blockage of the pump suction lines. A second concern, potential vortex formation, is not considered a serious concern for Mark I containment due to the large depth of the pool (approximately 25 feet) and the low approach velocities.

2. Justification for Continued Operation

With regard to potential blockage of the intake lines, the likelihood of any insulation being drawn into a emergency core cooling system pump suction line is very small. The potential debris in the drywell could only be swept into the suppression pool via the downcomer piping. However, the downcomer pipes (approximately two feet in diameter) are capped with jet deflectors and would prevent any large pieces from reaching the suppression pool. Any smaller pieces reaching the pool would tend to settle on the bottom and would not be drawn into the pump suction since it is located several feet above the pool bottom. In addition, boiling water reactor designs employ strainers within the suction piping, and net positive suction head calculations for the pump are based on an assumed 50 percent blockage.

Accordingly, we conclude that Dresden can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public.

TASK A-44 Station Blackout

1. Description of Problem

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

Task A-44 involves a study of whether or not nuclear power plants should be designed to accommodate a complete loss of all alternating current power, i.e., a loss of both the offsite and the emergency diesel generator alternating current power supplies. This issue arose because of operating experience regarding the reliability of alternating current power supplies. A number of operating plants have experienced a total loss of offsite electrical power, and more occurrences are expected in the future. During each of these loss-of-offsite power events, the onsite emergency alternating current power supplies were available to supply the power needed by vital safety equipment. However, in some instances, one of the redundant emergency power supplies has been unavailable. In addition, there have been numerous reports of emergency diesel-generators failing to start and run in operating plants during periodic surveillance tests.

2. Justification for Continued Operation

A loss of all alternating current power was not a design basis event for the Dresden Unit 2 facility. However, there are a number of items that assure safe continued operation.

First, Dresden Unit 2 Auxiliary Power System is connected to a highly reliable power grid. Sources of offsite power are available from both the 345KV and 138KV systems through various switching combinations. To date, Dresden 2 has never experienced a total loss of offsite power and the probability of such an event has been shown to be extremely low.

Second, if by some remote chance, offsite alternating current power is lost, three diesel-generators and their associated distributed systems will deliver emergency power to safety-related equipment. Historical records have proven that these diesels are highly reliable and operational experience has proven this fact. Maintenance and surveillance procedures are currently in effect which will assure that this high reliability is maintained at all times.

Third, if both offsite and onsite alternating current power are lost, the isolation condenser and HPCI may be used to remove core decay heat without reliance on alternating current power. This will assure that adequate cooling can be maintained during the brief time period until either offsite or onsite power sources are restored.

Finally, specific Blackout Training Procedures have been implemented to assure that adequate core cooling will be maintained and that power will be restored in a timely manner.

Based on the above, CECo has concluded that there is reasonable assurance that Dresden Unit 2 can be operated prior to the ultimate resolution of this generic issue without endangering the health and safety of the public.

Task A-45 Shutdown Decay Heat Removal Requirements

1. Description of Problem

   Following a reactor shutdown, the radioactive decay of fission products continues to produce heat (decay heat) which must be removed from the primary sytem. The alternative means of core cooling will be evaluated to determine the adequacy of the decay heat removal systems at Dresden Unit 2.

2. Justification for Continued Operation

   The principal means for removing heat in a boiling water reactor while at high pressure is via the steam lines to the turbine condenser. The condensate is normally returned to the reactor vessel by the feedwater system. However, the isolation condenser is provided for core cooling in the event that the reactor becomes isolated from the main condenser. The high pressure coolant injection system can provide makeup and cooling. Both the isolation condenser and the high pressure coolant injection system have redundant cooling water systems.

   If the isolation condenser and the high pressure coolant injection system are unavailable, the reactor system pressure can be reduced by the automatic depressurization system so that cooling by the low pressure coolant injection system and core spray systems can be initiated. The heat rejected to the suppression pool is removed through the low pressure coolant injection system heat exchangers which is cooled by the containment cooling service water.

   The normal mode of shutdown is to remove decay heat through the shutdown cooling system. The function of this system is to cool and maintain primary water temperature at 125^{o}F.

   Dresden Unit 2 has a dedicated diesel generator which has the capability of providing power to the equipment for Engineered Safety Systems Division II. A swing diesel generator will provide power to Engineered Safety Systems Division I. The capability has been provided, as an additional safety feature, for the Unit 3 diesel generator to provide power to the Unit 2 diesel generator's bus through a bus tie joined by two normally open circuit breakers.

   Based on the above, we conclude that Dresden Unit 2 can be operated prior to ultimate resolution of this generic issue without endangering the health and safety of the public.

TASK A-46 Seismic Qualification of Equipment in Operating Plants

1. Description of Problem

The design criteria and methods for the seismic qualification of mechanical and electrical equipment in nuclear power plants have undergone significant change during the course of the commercial nuclear power program. Consequently, the margins of safety provided in existing equipment to resist seismically induced loads and perform the intended safety functions may vary considerably. The seismic qualification of the equipment in operating plants must, therefore, be reassessed to ensure the ability to bring the plant to a safe shutdown condition when subject to a seismic event. The objective of this unresolved Safety Issue is to establish an explicit set of guidelines that could be used to judge the adequacy of the seismic qualification of mechanical and electrical equipment at all operating plants in lieu of attempting to backfit current design criteria for new plants. This guidance will concern equipment required to safely shutdown the plant, as well as equipment whose function is not required for safe shutdown, but whose failure could result in adverse conditions which might impair shutdown functions.

2. Justification for Continued Operation

Although many operating plants were designed before the development of current licensing criteria, the design rules and procedures incorporated inherent conservatisms. These include: (1) the margins between allowable stresses and ultimate strength of engineering materials, (2) the methods used for combining loads, (3) the inherent ducitility of materials, and (4) the seismic resistance of nonstructural elements which are not normally considered in design calculations.

An expanding data base of observations at large industrial facilities that have experienced strong ground motion suggests that these facilities have significant seismic resistance capabilities. From the data, it can be concluded that the inherent seismic resistance of engineered structures and equipment is usually much greater than is assumed in both past and current analysis and design procedures. Even facilities designed with very nominal seismic considerations, have been able to withstand severe seismic environments without loss of safety function. When even the most modest attention is paid in design to providing lateral load carrying paths, significant capability results. Nuclear power plants have been designed using more vigorous techniques; therefore, it is reasonable to expect even higher inherent margins than are implied from the data base of observations. Because of the experience gained in the reivew of the SEP facilities and the continued staff review of seismic issues, it is concluded that operating plants can continue to operate without endangering the health and safety of the public, pending resolution of this Unresolved Safety Issue.

Dresden Unit 2 has also been seismically reviewed by the NRC's SEP program. The report on this review was published as NUREG/CR-0891. The review concluded in Section 2.4 that in general Dresden Unit 2 could adequately resist an earthquake with a SSE value of acceleration of 0.2 g. Although this conclusion is predicated on certain assumptions, the following items support or make conservative this conclusion:

1. The NRC has determined through the SEP Program that the Dresden 2 ground acceleration is 0.13 g- this is 35% less than the NUREG document found acceptable. (Reference letter from D.C. Crutchfield to SEP Owners, June 17, 1981.)

2. In response to SEP findings (I.E. Information Notice 80-21) electrical equipment anchorages have been reviewed and upgraded as required.

3. In response to I.E. Bulletin 79-14 the "as-built safety related piping systems" are being reanalyzed and resupported as required.

3. Corrective Action

In response to SEP topic III-6 and the NRC's Senior Seismic Reivew Team's report on Dresden Unit 2, CECo has initiated several actions to address NRC concerns. They are:

1. A thorough review was conducted by CECo of electrical equipment anchorage. This included a field walkdown of the anchorage of such equipment , seismic analyses and modifications to the anchorages if they were determined to provide less than the desired margin of safety.

2. The wooden bracing on the battery racks will be replaced by angle iron to improve the margin of safety during seismic events.

3. CECo is involved in a SEP Owner Group with J.A. Blume & Associates to investigate the seismic adequacy of rod hung cable trays.

4. CECo is also involved in an owners group which is presently investigating the response of equipment which has been through and earthquake.

5. Response to I.E. Bulletin 79-14 has resulted in substantial reanalysis and modifications to supports for "as-built" safety related piping systems. This work is presently in progress.

TASK A-47 Safety Implications of Control Systems

1. Description of Problem

This issue concerns the potential for transients or accidents being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration. One concern is the potential for a single failure such as a loss of power supply, short circuit, open circuit, or sensor failure to cause simultaneous malfunction of several control features. Such an occurrence would conceivably result in a transient more severe than those transients analyzed as anticipated operational occurrences. A second concern is for a postulated accident to cause control system failures which would make the accident more severe than analyzed. Accidents could conceivably cause control system failures by creating a harsh environment in the area of the control equipment or by physically damaging the control equipment. Although it is generally believed that such control system failures would not lead to serious events or result in conditions that safety systems cannot safely handle, in-depth studies have not been rigorously performed to verify this belief. The purpose of this "Unresolved Safety Issue" is to define generic criteria that will be used for plant-specific review.

2. Justification for Continued Operation

The Dresden Unit 2 control and safety systems have been designed with the goal of ensuring that control system failures (either single or multiple failures) will not prevent automatic or manual initiation and operation of any safety system equipment required to trip the plant or to maintain the plant in a safe shutdown condition following any "anticipated operational occurrence" or "accident". This has been accomplished by either providing independence between safety and nonsafety systems or providing isolating devices between safety and nonsafety systems. These devices preclude the propagation of nonsafety system equipment faults such that operation of the safety equipment is not impaired.

A systematic evaluation of the control system design, such as contemplated for this "Unresolved Safety Issue," has not been performed to determine whether postulated accidents could cause significant control system failures which would make the accident consequences more severe than presently analyzed. However, operating experience of more than 40 reactor years has shown that control system failures can be corrected by operator or automatic actions. To date, no control system failure has resulted in a significant safety hazard.

Based on the above, CECo has concluded that there is reasonable assurance that Dresden Unit 2 can be operated prior to the ultimate resolution of this generic issue without endangering the health and safety of the public.

TASK A-48 Hydrogen Control Measures and Effects Of Hydrogen Burns on Safety Equipment

1. Description of Problem

Following a loss-of-coolant accident in a light water reactor plant, combustible gases, principally hydrogen can accumulate inside the primary reactor containment as a result of: (1) metal-water reaction involving the fuel element cladding; (2) the radioactive decomposition of the water in the reactor core and the containment sump; (3) the corrosion of certain construction materials by the spray solution; and (4) any synergistic chemical, thermal and radiolytic effects of post-accident environmental conditions on containment protective coating systems and electric cable insulation.

Because of the potential for significant generation as the result of an accident, 10 CFR Section 50.44, "Standards for Combustible Gas Control System in Light Water Cooled Power Reactors," and Criterion 41 of the General Design Criteria, "Containment Atmosphere Cleanup," in Appendix A to 10 CFR Part 50, require that systems be provided to control hydrogen concentrations in the containment atmosphere following a postulated accident to ensure that containment integrity is maintained.

On December 21, 1981, the NRC published a final ruling in the Federal Register concerning interim hydrogen control for BWR primary containments. This new ruling, 10 CFR 50.44(c)(3)(ii), requires that BWR plants with Mark I or Mark II containments relying on purge/repressurization as the primary means of combustible gas control be provided with either an internal hydrogen recombiner or be provided with the capability to install an external recombiner following an accident. This new ruling requires that modifications be completed by the end of the first scheduled refueling outage of sufficient duration beginning after July 15, 1982.

2. Justification of Continued Operation

The primary containment at Dresden Unit 2 relies on nitrogen inerting as the primary means of combustible gas control. In addition to the primary containment nitrogen inerting, an Air Containment Atmospheric Dilution (ACAD) system and a Containment Atmospheric Monitoring (CAM) system have also been installed to monitor and control hydrogen gas concentration in the primary containment during a postulated LOCA. It should be noted that a commitment has been made to upgrade the existing CAM system. This modification work is scheduled for completion by December, 1983.

Having technically reviewed the most current provisions stated in 10CFR50.44 (c)(3)(ii), it is the position of Commonwealth Edison that since the primary containment at Dresden Unit 2 relies on nitrogen inerting rather purge/repressurization (ACAD) as a primary means of combustible gas control, installation of hydrogen recombiners is not required. Technical justification for this position is based on the results of a recent analysis performed by the General Electric Company, "Generation and Mitigation of Combustible Gas Mixtures in

Inerted BWR Mark I Containments" (NEDO-22155). Results of the analysis performed show that following a postulated LOCA, peak oxygen concentrations found within a BWR Mark I containment would remain below the combustible gas limits at all times without the need for containment venting or hydrogen recombiners. The results of the analysis also show that an ACAD system is not required. In the event of a postulated LOCA, ACAD would further pressurize the drywell, and without controlled venting, would not effectively control combustible gases.

With regard to the results of this analysis and in consideration of the commitment to inert the Dresden Unit 2 primary containment with nitrogen, we conclude that Dresden Unit 2 can continue to operate without undue risk to the health and safety of the public.