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REGULATORY DOCKET FILE COPY

October 13, 1976

DOCKETED USNRC OCT 15 1976 U.S. NUCLEAR REGULATORY COMMISSION Mail Section

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Mr. Benard C. Rusche, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Dresden Station Unit 3 - Proposed Amendment to Technical Specifications for DPR-25 to Maintain Structural Integrity in Accordance with the 1974 Edition of the ASME Boiler and Pressure Vessel Code - NRC Docket No. 50-249

Reference (a): Amendment No. 13 to Dresden Unit 1, Facility Operating License No. DPR-2 Issued January 23, 1976, NRC Docket No. 50-10.

Dear Mr. Rusche:

Pursuant to 10 CFR 50.59, Commonwealth Edison proposes to amend Section 3.6.F of Appendix A Technical Specifications to Facility Operating License No. DPR-25 for Dresden Station Unit 3. The proposed amendment will permit the maintenance of the reactor vessel integrity in accordance with the provisions of the 1974 Edition Summer 1975 Addenda of the ASME Boiler and Pressure Vessel Code. The inservice inspection program based on the 1971 edition is unchanged by this request.

The 1974 edition with later addenda provide guidance for evaluating indications discovered during the inservice inspections.

The change is needed in order to evaluate indications discovered during a recent ultrasonic test inspection of the reactor vessel feedwater nozzles. The original acceptance criterion contained no provisions for evaluation of such indications and the current specification requires structural integrity be maintained in accordance with the original acceptance criterion. The proposed change is nearly identical to Reference (a) previously reviewed and approved by your staff. The enclosed amended pages contain the proposed revisions.

The proposed changes have received on-site and off-site review.

Three (3) signed originals and 37 copies are provided for your use.

SUBSCRIBED and SWORN to before me this 13th day of October, 1976.

Nancy M. Hollingsworth
Notary Public

Very truly yours,

R H Bolger

R. L. Bolger
Assistant Vice President

10449

Enclosure (1): 40 Copies of Amended Pages iii, 91, 91a, 91b, 98, and 98a for Appendix A Technical Specifications for DPR-25.

F. Structural Integrity

The structural integrity of the primary system boundary shall be maintained at the level required by the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", 1974 Edition, **Summer 1975 Addenda. (ASME) Code Section XI).** Components of the primary system boundary whose inservice examination reveals the absence of flaw indications or flaw indications not in excess of the allowable indication standards of this Code are acceptable for continued service. Plant operation with components which have inservice examination flaw indication(s) in excess of the allowable indication standards of the Code shall be subject to NRC approval.

a. Components whose inservice examination reveals flaw indication(s) in excess of the allowable indication standards of the ASME Code, Section XI, are unacceptable for continued service unless the following requirements are met:

F. STRUCTURAL INTEGRITY

The nondestructive inspections listed in Table 4.6.1 shall be performed as specified in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition, Summer 1971 Addenda. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions of this evaluation will be reviewed with the NRC.

(i) An analysis and evaluation of the detected flaw indication(s) shall be submitted to the NRC that demonstrate that the component structural integrity justifies continued service. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications", of ASME Code, Section XI.

(ii) Prior to the resumption of service, the NRC shall review the analysis and evaluation and either approve resumption of plant operation with the affected component or require that the component be repaired or replaced.

b. For components approved for continued service in accordance with paragraph a. above, reexamination of the area containing the flaw indication(s) shall be conducted during each scheduled successive inservice inspection. An analysis and evaluation shall be submitted to the NRC following each inservice inspection. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications", of ASME Code, Section XI, and shall reference prior analyses submitted to the NRC to the extent applicable. Prior to resumption of service following each inservice inspection, the NRC shall review the analysis and evaluation and either approve resumption of plant operation with the affected component or require that the component be repaired or replaced.

c. Repair or replacement of components, including reexaminations, shall conform with the requirements of the ASME Code, Section XI. In the case of repairs, flaws shall be either removed or repaired to the extent necessary to meet the allowable indication standards specified in ASME Code, Section XI.

G. Jet Pumps

1. Whenever the Reactor is in the Startup/Hot Standby or Run modes, all jet pumps shall be intact and all operating jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

2. Flow indication from each of the twenty jet pumps shall be verified prior to initiation of reactor startup from a cold shutdown condition.

3. The indicated core flow is the sum of the flow indication from each of the twenty jet pumps. If flow indication failure occurs for two or more jet pumps, immediate corrective action shall be taken. If flow indication for all but one jet pump cannot be obtained within 12 hours an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

G. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the Startup/Hot Standby or Run modes, jet pump integrity and operability shall be checked daily by verifying that the following two conditions do not occur simultaneously:

   a. The recirculation pump flow differs by more than 10% from the established speed-flow characteristics.

   b. The indicated total core flow is more than 10% greater than the core flow value derived from established power-core flow relationships.

2. Additionally, when operating with one recirculation pump with the equalizer valves closed, the diffuser to lower plenum differential pressure shall be checked daily, and the differential pressure of any jet pump in the idle loop shall not vary by more than 10% from established patterns.

The performance of reactor coolant leakage detection system will be evaluated during the first five years of station operation and the conclusions of this evaluation will be reported to the AEC.

It is estimated that the main steam line tunnel leakage detection system is capable of detecting of the order of 3000 lb/hr. The system performance will be evaluated during the first five years of plant operation and the conclusions of the evaluation will be reported to the AEC.

E. Safety and Relief Valves — Experience in safety valve operation shows that a testing of 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. The tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as ±1% of design pressure. An analysis has been performed which shows that with all safety valves set 1% higher the reactor coolant pressure safety limit of 1375 psig is not exceeded. Solenoid actuated relief valves are used to avoid activation of the safety valves. In view of the fact that the solenoid activated relief valves are more complicated, it is prudent to test them at each refueling outage. The safety valves are required to be operable above the design pressure (90 psig) at which the core spray subsystems are not designed to deliver full flow.

F. Structural Integrity — A pre-service inspection of the components listed in Table 4.6.1 will be conducted after site erection to assure the system is free of gross defects and as a reference base for later inspections. Prior to operation, the reactor primary system will be free of gross defects. In addition, the facility has been designed such that gross defects should not occur throughout life. The inspection program given in Table 4.6.1 was based on Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition, Summer 1971 Addenda, which was followed except where accessibility for inspection was not provided. This edition of the Code is suitable for detecting flaw indications but does not provide adequate guidance for the evaluation of ultrasonic reflectors. The requirement in the 1971 Edition of Section XI that the operator evaluate the reflector to determine the size, shape, and nature can best be satisfied by examination and evaluation of the flaw in accordance with the techniques presented in Appendix A to ASME Section XI in the 1974 Edition, Summer 1975 addenda. It is the intent of this specification to require inservice inspection of the primary system boundary per Table 4.6.1 of this specification and the 1971 Edition of ASME Section XI including the Summer 1971 Addenda and to permit the evaluation of flaws in excess of the acceptance standards of that Edition and Addenda in accordance with the techniques of the 1974 version. Commonwealth Edison Company recognizes the importance of inspection of those areas which are presently not accessible and will study and implement, if practicable, new means to include those areas within the inspection program. This inspection provides further assurance that gross defects are not occurring after the system is in service. This inspection will reveal problem areas should they occur before a leak develops.

The special inspection of the main feed and steam lines is to provide added protection against pipe whip. The GRP I welds are selected on the basis of an analysis that shows these welds are the highest stress welds and that due to their physical location, a break would result in the least interference and maximum energy upon impact with the drywell. These welds are the only ones which offer any significant risk and are therefore inspected four times as often as the other welds within the drywells.

GRP II welds are selected because without regard for the operating stress levels and interfering equipment, they have sufficient theoretical energy to penetrate and would propel the pipe toward the containment. They are therefore included in first inspection. Upon consideration of impact angle, interfering equipment and distance pipe travels, no substantial risk is involved and no extra inspection is needed.

In addition, extensive visual inspection for leaks will be made periodically on critical systems. The inspection program specified encompasses the major areas of the vessel and piping systems within the drywell. The inspection period is based on the observed rate of growth of defects from fatigue studies sponsored by the AEC.