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Regulatory File Cy.

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RECEIVED MAR 1 1974 U.S. ATOMIC ENERGY COMMISSION Regulatory Mail Section

Mr. J. F. O'Leary, Director
Directorate of Licensing
Office of Regulation
U.S. Atomic Energy Commission
Washington, D.C. 20545

Subject: Proposed Change to Appendix A, DPR-25
AEC Dkt 50-249

Dear Mr. O'Leary:

Pursuant to Section 50.59 of 10CFR50 and Paragraph 3.B of Facility Operating License DPR-25, Commonwealth Edison Company hereby submits a proposed change to Appendix A of DPR-25 (Dresden Unit 3). The purpose of this change is to modify the Technical Specifications concerning the Inservice Inspection Program which is delineated in Table 4.6.1 of the Technical Specifications. The page changes to the Technical Specifications are attached and the reason for the proposed change is given below.

When the Technical Specifications for Dresden Unit 3 were developed, the Inservice Inspection Program utilized drafts of Section XI of the ASME Boiler and Pressure Vessel Code. The changes made in the attached pages are designed to incorporate revisions to the examination categories made to the draft of the Code, by either the finalized version dated July 1, 1971 or Addenda to it. This change would help standardize and update the Dresden Unit 3 Technical Specifications to those of Dresden Unit 2, Quad-Cities 1 and 2, and the latest code examination categories. You will note that the proposed change converts from volumetric to visual the examination requirements for the CRD housing to vessel welds and that this change provides the same result as previously approved by the Atomic Energy Commission for Dresden Unit 2 on January 8, 1973. Additionally a requirement to volumetrically examine the control rod drive housings pressure retaining welds has been added.

This proposed change has been reviewed and approved by Onsite and Offsite Review.

Three (3) signed originals and 37 copies of this proposed change are submitted for your review.

Very truly yours,

R. Tuetken

for/ J. S. Abel
Nuclear Licensing Administrator
Boiling Water Reactors

Att.

SUBSCRIBED and SWORN to
before me this 27th day
of February, 1974.

Brenda Pammer
Notary Public

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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, Rules for Inservice Inspection of Nuclear Reactor Coolant Systems, which was followed except where accessibility for inspection was not provided. The 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.

TABLE 4.6.1 (cont)

Category	Examination Area	Exam Method	Inspection Interval	Extent of Examinations
2)	Nozzle-to-vessel head welds & nozzle-to head inside radiused section	Visual and Volumetric	Cumulative 100% coverage of nozzle-to-shell weld, & 100% of inner radius section of the nozzle-to-shell juncture	2) Nozzle-to-vessel Head Welds Head Instrumentation (2) Head Spray Inlet (1)
E	**Partial penetration welds including control rod drive penetrations and vessel instrumentation nozzles**	**Visual**	**The examinations performed during each inspection interval shall cover at least 25% of each group of penetrations of comparable size and function**	**The area surrounding each penetration shall be examined for evidence of leakage during pressure test.**
F 1)	Primary Nozzles to safe-end welds	Visual, Surface, & Volumetric	Cumulative 100% at end of 10 year interval	1) Safe-ended nozzles Recirc. Outlet (2) - 1/5 years Recirc. Inlet (10) - 1/year Main Steam Outlet (4) - 1/2.5 years Feedwater Inlet (4) - 1/2.5 years Isolation Condenser Outlet (2) 1/5 years Core Spray Inlet (2) - 1/5 years Control Rod Drive Return (1) 1/10 years Liquid Poison (1) - 1/10 years
2)	Vessel, Pump & Valve safe-ends to primary pipe welds & safe-ends in branch piping welds	Visual, Surface, & Volumetric	Cumulative 100% at end of 10 year interval	2) All safe-end welds in pump, valve & branch piping to be inspected within the 10 year interval.

TABLE 4.6.1 (Cont)

Category	Examination Area	Exam Method	Inspection Interval	Extent of Examination
M 1)	Welds in valve bodies 3" and above	Visual and Volumetric	One valve of each type during 10-year interval	Not applicable with present plant design
2)	Valve bodies 3" and above	Visual	One valve of each type during 10-year interval	One disassembled valve (with or without welds and 3" over in normal size) in each category and type shall be subject to visual examination. Individual examinations shall cover 100% of the pressure boundary and may be performed at or near the end of the 10 year interval.
N	Interior surfaces and internals and integrally welded internal supports of the reactor vessel, including core spray spargers, core spray nozzles, and upper portions of jet pumps	Visual	During first refueling outage and during subsequent refueling outages at approximately 3-year intervals	Interior surfaces and internal components of the reactor vessel, including the space at the bottom head, and internal attachments which are welded to the vessel, made accessible by the removal of components during normal refueling operations. All internal attachments whose failure may adversely affect core integrity shall be examined.

TABLE 4.6.1 (Cont)

Category	Examination Area	Exam Method	Inspection Interval	Extent of Examination
O	Control rod drive housings pressure-retaining welds	Volumetric	The examinations performed during each inspection interval shall include the welds in 10% of the peripheral control rod drive housings.	The areas shall include the weld metal and base metal for one wall thickness beyond the edge of the weld.

Supplemental Inspection Program for
First and Second Refueling Outages

(a) The following critical and sensitized components shall be nondestructively examined by the methods indicated:

Component	Examination Method
1. Field clad-repaired safe-ends	PT and UT
2. Bimetallic welds of field-replaced safe-ends	PT and (UT or RT)
3. Field clad-repaired sensitized components within reactor vessel whose failure could adversely affect safety	Visual

(b) The areas subject to examination shall include 100 percent of the exterior surfaces of clad repaired safe-ends (a) 1, and 100 percent of the exterior surfaces of welds (a) 2. Weld areas to be examined shall include the base material for, at least, one wall thickness beyond the edge of weld.