June 29, 1982

Docket No. 50-237
LS05-82-06-116

Mr. L. DelGeorge
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. DelGeorge:

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNIT 2 - SEP TOPIC III-8.C, IRRADIATION DAMAGE, USE OF SENSITIZED STAINLESS STEEL AND FATIGUE RESISTANCE

Reference: Letter, R. F. Janacek to D. L. Ziemann, SEP Topic III-8.C, dated March 10, 1980

Enclosed is a copy of our final evaluation of Systematic Evaluation Program Topic III-8.C, "Irradiation Damage, Use of Sensitized Stainless Steel and Fatigue Resistance."

This assessment compares your facility as described in Docket No. 50-237, with the criteria currently used by the regulatory staff for licensing new facilities. The final evaluation differs from the draft in that it incorporates your comment, has been rewritten into a new format and has been reworded for clarity.

This evaluation will be a basic input to the integrated safety assessment for your facility. This topic assessment may be changed in the future if your facility design is changed or if NRC criteria relating to this topic are modified before the integrated assessment is completed.

Sincerely,

SE04

DSU USE (11)

Paul O'Connor, Project Manager
Operating Reactors Branch No. 5
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

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DATE	6/24/82	6/24/82	6/24/82	6/24/82	6/25/82	6/29/82	6/28/82

Mr. L. DelGeorge

cc

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SYSTEMATIC EVALUATION PROGRAM
TOPIC III-8.C

DRESDEN 2

TOPIC: III-8.C, IRRADIATION DAMAGE, USE OF SENSITIZED STAINLESS STEEL AND FAITGUE RESISTANCE

I. INTRODUCTION

The reactor internals are designed to support and orient the reactor core and control assemblies, provide a flow path for reactor coolant and support in-core instrumentation. The internals are to withstand the forces due to weight, pre-load of fuel assemblies, control rod dynamic loading, vibration, and loss of coolant accident blowdown coincident with earthquake accelerations.

SEP Topic III-8.C is intended to determine if the integrity of the reactor internal structures has been degraded through the use of sensitized steel.

The effect of neutron irradiation and fatigue resistance on material of the internal structures was eliminated from the safety objective of Topic III-8.C in memorandum to D. G. Eisenhut from D. K. Davis and V. S. Noonan dated December 8, 1978. The memorandum concluded that operating experience indicated that no significant degradation of the materials of the reactor internal structures had occurred as a result of either irradiation damage or fatigue resistance. Furthermore, the Standard Review Plan does not address neutron irradiation nor fatigue resistance of the materials of the structures.

II. REVIEW CRITERIA

General Design Criterion 4, "Environmental and Missile Design Bases," Appendix A, 10 CFR Part 50, requires that components be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accident conditions. The use of sensitized stainless steel in the presence of certain environmental conditions can lead to stress corrosion cracking and the eventual loss of structural integrity of the affected component.

III. RELATED SAFETY TOPICS

SEP Topics III-8.A and III-8.B evaluate related items such as control rod drive mechanism integrity and loose parts monitoring, respectively. SEP Topic V-4 evaluates the structural integrity of reactor coolant pressure boundary safe-ends that have been fabricated with furnace sensitized stainless steel.

IV. REVIEW GUIDELINES

The review of the use of sensitized stainless steel in reactor internals was conducted in accordance with the acceptance criteria of Section 4.5.2, "Reactor Internal and Core Support Materials," of the Standard Review Plan and Regulatory Guides 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," and 1.44, "Control of the Use of Sensitized Stainless Steel." The materials specifications requirements were those of Sections II and III of the ASME Boiler and Pressure Vessel Code.

V. EVALUATION

The reactor is described in Sections 3 and 4 of the Safety Analysis Report for the Dresden Nuclear Power Station Unit No. 2. The internal components were designed to provide support for the fuel and maintain structural clearances during normal and accident conditions. In addition, the internal components provide passageway for the coolant to cool the fuel and means for adequately separating the steam from the coolant water.

The vessel was designed, fabricated and tested in accordance with Section III of the ASME Boiler and Pressure Vessel Code, 1963 Edition, through Summer 1964 Addenda. The materials used for the construction of the reactor internals were identified as Type 304 and Type 308 stainless steel, Inconel, and minor quantities of special purpose alloys. The identified structural materials have been used on other General Electric designed reactors and have proven adequate by the results of extensive tests, prior usage and satisfactory performance.

The regulatory position on the use of sensitized stainless steel in reactor internal materials was not addressed in the Safety Analysis Report for the Dresden Nuclear Power Station Unit No. 2. Experience has shown that at least three elements in combination are necessary to cause cracking in sensitized stainless steel components. These are material susceptibility, an oxygenated water environment, and a threshold total stress. We have not ascertained for this evaluation that the Dresden Unit No. 2 reactor internal components contain sensitized stainless steel in contact with an oxygen saturated coolant water environment. However, the calculated stresses on the reactor internal components do not exceed the threshold stress values associated with intergranular stress corrosion cracking. The threshold stress values are less than the 0.2% off-set yield stress at temperature. Further, in the reactor environment, stress relaxation may occur due to irradiation and temperature effects.

The Licensee Event Reports and the BWR Nuclear Power Experience were reviewed for the Dresden Nuclear Power Station Unit No. 2 in order to correlate reactor internal materials failure to the use of sensitized stainless steel in the components. We did not find any incidents of sensitized stainless steel associated with stress corrosion cracking of the reactor internals.

We conclude from our review of the Licensee Event Reports and the BWR Nuclear Power Experience that the integrity of the reactor internal components was not degraded by the use of sensitized stainless steel.

The inservice inspection program for the reactor internal components is being conducted during the current interval to the requirements of Section XI of the ASME Boiler and Pressure Vessel Code, 1977 Edition, including Summer 1975 Addenda. The program is in compliance with paragraph (g) of Section 50.55a of 10 CFR Part 50. It will assure that the integrity of the components is maintained during reactor operation.

VI. CONCLUSIONS

We conclude from our review of the information submitted by the licensee that the materials in the reactor internal components are probably not sensitized and are operated in an oxygen saturated water environment, and that the incidents of stress corrosion cracking are rare because the total stress level is relatively low, not exceeding the 0.2% off-set yield strength at operating temperature. In the unlikely event that intergranular stress corrosion cracking should occur, cracks in the components will be detected by inservice inspection procedures prior to component failure. We conclude that the integrity of the reactor internal components will be assured by the inservice inspection program conducted to the requirements of Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition, including Summer 1979 Addenda, in compliance with paragraph (g) of Section 50.55a of 10 CFR Part 50. Further, we conclude that intergranular stress corrosion cracking in the reactor internal components is not a hazard to the health and safety of the public.