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Docket No. 50-237

JAN 17 1980

Mr. D. Louis Peoples
Director of Nuclear Licensing
Commonwealth Edison Company
P. O. Box 767
Chicago, Illinois 60690

Dear Mr. Peoples:

RE: SEP TOPIC III-8.C Irradiation Damage, Use of Sensitized Stainless Steel and Fatigue Resistance

Enclosed is a copy of our draft evaluation of Systematic Evaluation Program Topic III-8.C. You are requested to examine the facts upon which the staff has based its evaluation and respond either by confirming that the facts are correct, or by identifying any errors. If in error, please supply corrected information for the docket. We encourage you to supply for the docket any other material related to these topics that might affect the staff's evaluation.

Your response within 30 days of the date you receive this letter is requested. If no response is received within that time, we will assume that you have no comments or corrections.

Sincerely,

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosure:
Topics III-8.C

cc w/enclosure:
See next page

OFFICE	DOR: ORB #2	DOR: ORB #2	DOR: ORB #2			
SURNAME	PO'Connor:cc	HSmith	DLZiemann		8002110	038
DATE	1/16/80	1/16/80	1/17/80			

cc
Isham, Lincoln & Beale
Counselors at Law
One First National Plaza, 42nd Floor
Chicago, Illinois 60603

Mr. B. B. Stephenson
Plant Superintendent
Dresden Nuclear Power Station
Rural Route #1
Morris, Illinois 60450

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N. W.
Washington, D. C. 20005

U. S. Nuclear Regulatory Commission
ATTN: Jimmy L. Barker
P. O. Box 706
Morris, Illinois 60450

Susan N. Sekuler
Assistant Attorney General
Environmental Control Division
188 W. Randolph Street
Suite 2315
Chicago, Illinois 60601

Morris Public Library
604 Liberty Street
Morris, Illinois 60451

Chairman
Board of Supervisors of
Grundy County
Grundy County Courthouse
Morris, Illinois 60450

Department of Public Health
ATTN: Chief, Division of
Nuclear Safety
535 West Jefferson
Springfield, Illinois 62761

Director, Technical Assessment
Division
Office of Radiation Programs
(AW-459)
U. S. Environmental Protection
Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection
Agency
Federal Activities Branch
Region V Office
ATTN: EIS COORDINATOR
230 South Dearborn Street
Chicago, Illinois 60604

Mr. Neil Smith
Commonwealth Edison Company
P.O. Box 767
Chicago, Illinois 60690

SYSTEMATIC EVALUATION PROGRAM
PLANT SYSTEMS/MATERIALS
DRESDEN NUCLEAR POWER STATION UNIT NO. 2

Topic III-8.C - Irradiation Damage, Use of Sensitized Stainless Steel and Fatigue Resistance

The safety objective of this review is to determine whether the integrity of the internal structures of operating reactors has been degraded through the use of sensitized stainless steel.

The effect of neutron irradiation and fatigue resistance on materials 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 or fatigue. Furthermore, the Standard Review Plan (Section 4.5.2) does not address neutron irradiation nor fatigue resistance of the materials of the reactor internal structures.

As a result of incidents of intergranular stress corrosion cracking in piping in the BWR system, special study groups were formed by NRC and industry to evaluate the cause, extent and safety implications of the use of sensitized stainless steel in the nuclear steam supply systems. The study groups identified the incidents with the recirculation system bypass lines, the core spray lines, and the control rod drive return lines. It was concluded that the problem was caused by a combination of high total stresses, sensitization of the austenitic stainless steel in the heat affected zones of welds, and the relatively high oxygen content of the coolant.

The NRC study group recommended an augmented inservice inspection program for stainless steel piping, more stringent monitoring of the leak detection system, modification of plant operating practice, and the use of alternate materials immune to intergranular stress corrosion cracking. The study group concluded in NUREG-0531, "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants," that intergranular stress corrosion cracking in piping would be detected prior to unstable crack growth because of the adequacy of the inservice inspection program and the leak detection system. Reactor operating experience has validated the leak-before-break concept of piping integrity, and, it was concluded, that through-wall cracks in the piping systems would be detected before they presented a hazard to the health and safety of the public.

The regulatory position on the use of sensitized stainless steel in reactor internal materials is addressed in the Standard Review Plan Section 4.5.2, "Reactor Internal Materials." The areas currently reviewed in the applicant's SAR are materials specification and the controls imposed on the reactor coolant chemistry, fabrication practices and examination and protection procedures.

The materials specification should comply with Section III of the ASME Boiler and Pressure Vessel Code and the components should satisfy the recommendations of Regulatory Guide 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal" and Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel".

The reactor is described in Sections 3 and 4 of the Safety Analysis Report for the Dresden Nuclear Power Station Unit Nos. 2 and 3. 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, including 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 assume 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 near or greater 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. The events are summarized as follows:

> In LER's beginning in September 1974, and continuing through 1977, leaks were reported in the 4-in recirculation bypass line, the 10-in core spray lines, the 14-in HPCI lines, and the control rod drive penetration. The cause of these failures was intergranular stress corrosion cracking, resulting from either furnace or weld sensitization. The defective material was replaced with material immune to sensitization. An augmented inservice inspection program was conducted to reduce the consequence of the failure incidents in the affected piping.

In December 1974, cracks in the feedwater spargers were discovered. This problem is an item under review by Generic Technical Activity A-10, "BWR Feedwater Nozzle Cracking".

During inservice inspection of reactor internals in March 1976, loose restrainer clamp bolt keepers were found on the jet pumps. The cause of the failure was fatigue cracking of the tack-welded restrainer assembly. The loose keepers were rewelded to the original specification.

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 degraded by the use of sensitized stainless steel. The reported events were detected by the inservice inspection and testing program. The events were considered by the NRC Pipe Crack Study Group in NUREG-0531, "Investigation and Evaluation of Stress Corrosion Grading in Piping of Light Water Reactor Plants" and in the Generic Technical Activity A-10, "BWR Feedwater Nozzle Cracking".

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, 1974 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.

We conclude from our review of the information submitted by the licensee that the materials in the reactor internal components are 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% offset 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 1975 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.

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JAN 11 1980

Docket Nos. 50-107237

Mr. D. Louis Peoples
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Peoples:

In response to our letter of September 6, 1978, you provided information on your 1977 experience of control rods failing to fully insert. We evaluated your response along with those provided by other BWR owners. We concluded that such events are general and that the frequencies of occurrence vary among the BWRs, apparently because of different maintenance programs.

Our September 6, 1978 letter recommended that you maintain an ongoing tabulation of any additional such events. To determine whether or not the frequency of these events has changed for a given plant, we now request that you report your more recent experience. This report should include the experience from the time of the last reported occurrence through 1979. In addition, please provide a summary of other control rod drive malfunctions, such as unlatching, for the same time period.

The information should be provided within 90 days. For each event, identify the number of rods not fully inserted, the position of the rods, the cause for failure to fully insert, and any related maintenance activities.

This request for additional generic information is in accordance with the GAO blanket clearance number B-180225 (R0536) which expires June 30, 1981.

Sincerely,

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

cc: See next page

OFFICE	DOR:ORB #2	DOR:ORB #2				
SURNAME	SJNowicki:ah	DLZiemann				
DATE	1/10/80	1/11/80				

8002040154

cc
Isham, Lincoln & Beale
Counselors at Law
One First National Plaza, 42nd Floor
Chicago, Illinois 60603

Mr. B. B. Stephenson
Plant Superintendent
Dresden Nuclear Power Station
Rural Route #1
Morris, Illinois 60450

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N. W.
Washington, D. C. 20005

U. S. Nuclear Regulatory Commission
ATTN: Jimmy L. Barker
P. O. Box 706
Morris, Illinois 60450

Susan N. Sekuler
Assistant Attorney General
Environmental Control Division
188 W. Randolph Street
Suite 2315
Chicago, Illinois 60601

Morris Public Library
604 Liberty Street
Morris, Illinois 60451

Chairman
Board of Supervisors of
Grundy County
Grundy County Courthouse
Morris, Illinois 60450

Department of Public Health
ATTN: Chief, Division of
Nuclear Safety
535 West Jefferson
Springfield, Illinois 62761

Director, Technical Assessment
Division
Office of Radiation Programs
(AW-459)
U. S. Environmental Protection
Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection
Agency
Federal Activities Branch
Region V Office
ATTN: EIS COORDINATOR
230 South Dearborn Street
Chicago, Illinois 60604