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DOCKET NUMBER 50-269/270/287

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TO:	FROM:	DATE OF DOCUMENT
Mr. Edson G. Case	Duke Pwr. Company Charlotte, North Carolina William O. Parker, Jr.	10/14/77
		DATE RECEIVED 10/19/77

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DESCRIPTION Re their 9-27-77 ltr

Consists of info re assurance that radioactive releases are maintained as low as reasonably achievable at Oconee.....

ENCLOSURE

PLANT NAME: Oconee Units 1-2-3

RJL 10/19/77 (2-P)

FOR ACTION/INFORMATION

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BRANCH CHIEF: (7)	Schwencer		

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DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

October 14, 1977

RECEIVED
OCT 19, 1977
U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

Mr. Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. A. Schwencer, Chief
Operating Reactor Branch #1

Reference: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

Dear Sir:

In my letter of September 27, 1977 actions were described that would assure that radioactive releases are maintained as low as reasonably achievable at Oconee. These actions were in response to your letter of August 4, 1977. The first action of this letter concerned the turbine buildings sumps and installed radiation monitors. It was stated, in part, that during periods of known secondary system contamination, the turbine building sumps would only be batch released with prior sampling. In this regard, it is our current intent to use either batch release with prior sampling or the sump radiation monitor to control the release of liquids from turbine building sumps. The following incorporate the action above and reiterate the other actions presented in my letter of September 27, 1977:

1. Radiation monitors have been installed in the two turbine building sumps. These monitors provide alarms in the control room of inadvertent radioactive release entering the turbine building sump or upon malfunction of the monitor. In order to ensure that inadvertent radioactive releases from the sump do not occur either the radiation monitors will be operable or the sumps will be sampled prior to batch release. Either measure in itself is considered reasonable and effective to prevent inadvertent releases of radioactive liquids from the turbine building sumps to the environment. In instances when batch releases are made, no dependence will be made on the radiation monitors.

2. A composite, flow proportional water sampler will be installed at the outfall of the oil collection basin by March 1, 1978. Weekly, a gamma isotopic analysis will be performed on the composite samples. In the event of an inadvertent radioactive liquid release to the basin, an additional composite sample will be drawn. This sample will be analyzed for Sr^{89} and Sr^{90} along with the samples taken on a monthly basis from the other low level activity radioactive waste tanks.

7729201 42

3. Procedures will be instituted by November 1, 1977 which will require two independent valve alignment checks prior to discharging radioactive spent secondary system demineralizer resins to the receiving tanks. This procedure will be followed whenever the activity in the secondary system indicates that there has been primary to secondary leaks, and continue until all activity from the secondary system has been removed.

Although your letter requested that the above proposals be incorporated into the Oconee Technical Specifications, we do not feel that this action is required. The equipment and procedures listed above are extensions of the defense in depth concept to prevent inadvertent liquid effluent releases. They support the objectives and specifications written in Oconee Technical Specification 3.9, Release of Liquid Radioactive Waste, and as such assist in the effective control of the release of radioactive liquid wastes from the station. Any revision to the Technical Specification incorporating the above items would duplicate objectives and specifications presently contained in Oconee Technical Specification 3.9.

Very truly yours,

William O. Parker, Jr.

William O. Parker, Jr
WOP/mlr

Docket

OCT 11 1977

Docket Nos. 50-269
50-270
and 50-287

Duke Power Company
ATTN: Mr. William O. Parker, Jr.
Vice President
Steam Production
Post Office Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Gentlemen:

By letter dated March 9, 1977, we requested that you determine if the individual performing the function of Radiation Protection Manager (RPM) at the Oconee Nuclear Station, meets the minimum qualifications of Regulatory Guide 1.89, September, 1975. We further stated that if the RPM is so qualified, you should propose a technical specification which states that the RPM shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975. On the other hand, if the present incumbent does not meet the minimum requirements of the guide, we requested that you advise us of this fact and provide a written commitment that the successor to the incumbent will be so qualified and that you will propose a technical specification to that effect at the time a successor enters that position.

By letter dated May 13, 1977, you responded to our request by taking exception to the provisions of Regulatory Guide (R.G.) 1.8. Your principal objections were that the RPM should not be required to have a bachelor's degree and an additional 5 years experience.

This letter is to advise you that R.G. 1.8 does not require the RPM to have a bachelor's degree. Rather, the Guide says that he shall have a bachelor's degree or the equivalent in a science or engineering subject. To provide clarification of this point, our definition of "equivalent" in the context of R.G. 1.8, is as follows:

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(a) 4 years of formal schooling in science or engineering,
(b) 4 years of applied radiation protection experience at a nuclear facility,
(c) 4 years of operational or technical experience/training in nuclear power, or
(d) any combination of the above totaling 4 years.

It should be noted that the above requirement is in addition to the requirement for five years of professional experience in applied radiation protection as specified in the Guide.

It is our position that the ANSI 18.1-1971 standard does not provide the appropriate qualifications required for the onsite RPM whose responsibility is to manage a radiation program with an annual man-rem budget such as that at Oconee, and, that the requirements of R.G. 1.8 are necessary for the RPM at the station to assure that exposures from normal operations, maintenance, etc., are maintained at levels that are as low as is reasonably achievable.

Accordingly, we reiterate our request that you adopt the provisions of R.G. 1.8 for any replacement of the current RPM in accordance with our letter dated March 9, 1977. Please respond within 45 days of receipt of this letter.

Sincerely,

\S\

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

cc: See next page

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cc: Mr. William L. Porter
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Washington, D.C. 20005

Oconee Public Library
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Walhalla, South Carolina 29691

OCT 4 1977

Docket Nos. 50-269
50-270
and 50-287

Duke Power Company
ATTN: Mr. William O. Parker, Jr.
Vice President
Steam Production
Post Office Box 2178
422 South Church Street
Charlotte, North Carolina 28242

Gentlemen:

RE: FRACTURE TOUGHNESS AND POTENTIAL FOR LAMELLAR TEARING OF STEAM GENERATOR AND REACTOR COOLANT PUMP SUPPORT MATERIALS

During the course of the licensing action for North Anna Power Station Unit Nos. 1 and 2, a number of questions were raised as to the potential for lamellar tearing1/ and low fracture toughness of the steam generator and reactor coolant pump support materials for that plant. Two different steel specifications (ASTM A36-70a and ASTM A572-70a) covered most of the material used for these supports. Toughness tests, not originally specified and not in the relevant ASTM specifications, were made on those heats for which excess material was available. The toughness of the A36 steel was found to be adequate, but the toughness of the A572 steel was relatively poor at an operating temperature of 80 F. In this case, the applicant has agreed to raise the temperature of the ASTM A572 beams in the steam generator supports to a minimum temperature of 225 F prior to reactor coolant system pressurization to levels above 1000 psig. Auxiliary electrical heat will be employed to supplement the heat derived from the reactor coolant loop as necessary to obtain the required operating temperature of the structures.

1/ Lamellar tearing is a cracking phenomenon which occurs beneath welds and is principally found in rolled steel plate fabrications. The tearing always lies within the parent plate, often outside the transformed (visible) heat-affected zone (HAZ) and is generally parallel to the weld fusion boundary. Lamellar tearing occurs at certain critical joints usually within large welded structures involving a high degree of stiffness and restraint. Restraint may be defined as a restriction of the movement of the various joint components that would normally occur as a result of expansion and contraction of weld metal and adjacent regions during welding. ("Lamellar Tearing in Welded Steel Fabrication", The Welding Institute).

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Since similar materials and designs have been used on other nuclear plants, the concerns raised on the supports for the North Anna plant may be applicable for other operating PWR plants. It is therefore necessary to reassess the fracture toughness and potential for lamellar tearing of the steam generator and reactor coolant pump support materials for all operating PWR plants.

We will require certain information to make the necessary reassessment of the steam generator and reactor coolant pump support materials for your plant; therefore, please provide the following information within sixty (60) days after receipt of this letter:

1. Provide engineering drawings of the steam generator and reactor coolant pump supports sufficient to show the geometry of all principal elements. Provide a listing of materials of construction.

2. Specify the detailed design loads used in the analysis and design of the supports. For each loading condition (normal, upset, emergency and faulted), provide the calculated maximum stress in each principal element of the support system and the corresponding allowable stresses.

3. Describe how all heavy section intersecting member weldments were designed to minimize restraint and lamellar tearing. Specify the actual section thicknesses in the structure and provide details of typical joint designs. State the maximum design stress used for the through-thickness direction of plates and elements of rolled shapes.

4. Specify the minimum operating temperature for the supports and describe the extent to which material temperatures have been measured at various points on the supports during the operation of the plant.

5. Specify all the materials used in the supports and the extent to which mill certificate data are available. Describe any supplemental requirements such as melting practice, toughness tests and through-thickness tests specified. Provide the results of all tests that may better define the properties of the materials used.

6. Describe the welding procedures and any special welding process requirements that were specified to minimize residual stress, weld and heat affected zone cracking and lamellar tearing of the base metal.

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7. Describe all inspections and non-destructive tests that were performed on the supports during their fabrication and installation, as well as any additional inspections that were performed during the life of the facility.

In addition to the information requested above, please provide your own evaluation of the fracture toughness of the steam generator and reactor coolant pump support materials for your plant. Please inform us within thirty (30) days after receipt of this letter of your schedule for providing us with your evaluation. This generic request was approved by GAO, B-180225 (R0072), clearance expires July 31, 1980. Approval was given under a blanket clearance specifically for identified generic problems.

Sincerely,

/s/ A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

cc: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Docket Nos. 50-269
50-270
and 50-287

OCT 4 1977

Duke Power Company
ATTN: Mr. William O. Parker, Jr.
Vice President
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A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

cc: See next page

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