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RECIP.NAME RECIPIENT AFFILIATION
REID,R.W. Operating Reactors Branch 4

SUBJECT: Forwards supplemental response to NUREG-0578 re design review of plant shielding & environ qualification of equipment for spaces sys for use in post-accident operations operations,Timely NRC response requested.

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WILLIAM O. PARKER, JR.
VICE PRESIDENT
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TELEPHONE: AREA 704
373-4083

March 7, 1980

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. R. W. Reid, Chief
Operating Reactors Branch No. 4

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

Dear Sir:

Please refer to my letter of February 29, 1980. This letter provides information which supplements my earlier letters provided in regard to implementation of NUREG-0578.

In accordance with NUREG-0578, several items were responded to with proposed actions which require NRC approval prior to implementation. Currently, no NRC response has been received on any of these items. Timely response on these items is desirable in order to preclude delays in procurement of equipment and installation beyond January 1, 1981.

Very truly yours,

William O. Parker, Jr. By [illegible]

RLG:scs

Attachment

A047
3/1

2.1.6.b Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces Systems Which May Be Used in Post-Accident Operations

Methodologies used in the initial Oconee shielding review were described in W. O. Parker's January 2, 1980 letter. The results were listed as nine areas of concern over potential personnel access problems. Since the January 2 submittal, work has been underway to refine the initial review calculations. This has largely resulted in "as-built" geometry models, as opposed to the simplified models used in the initial review. The methodologies that have been used are as described in the January 2 submittal. Design revisions for the areas of concern are listed below.

1. Control Room -
   The Oconee control rooms are located adjacent to their respective unit's mechanical penetration room. The routing of low pressure recirculation piping within the penetration rooms could result in control room personnel exposures in excess of GDC 19. With the Technical Support Center located inside the Control Room Complex, TSC personnel exposures could also exceed GDC 19. Permanently installed lead shielding will be located along control room walls adjacent to the penetration rooms in such a manner as to assure personnel exposures in the Control Room and TSC do not exceed GDC 19. Design drawings will be released by July 1 with installation anticipated to be complete by January 1, 1981.

2. Administration Building (Unit 1 LOCA) -
   Final calculations show that the Administration Building is accessible. Radiation gradients will exist throughout building. As a result, portions of the building actually meet GDC 19, while other portions are accessible infrequently. In the unlikely event of a LOCA, station HP personnel will determine existing radiation gradients within the building. No vital functions are controlled from or centered in the Administration Building. No access problems are anticipated if a LOCA occurs in Units 2 or 3. No design modifications are necessary.

3. Administration Building Annex (Unit 1 LOCA) -
   Final calculations show that the Administration Building Annex will not be accessible for approximately the first four days of a LOCA. After this period of time, the whole building will be accessible. No vital functions are controlled from or centered in the Administration Building Annex. No access problems are anticipated if a LOCA occurs in Units 2 or 3. No design modifications are necessary.

4. Counting Room (Unit 2 LOCA) -
   The counting room is accessible. However, the radiation levels will affect the sensitivity of the counting equipment if a LOCA were to occur in Unit 2. In this event, an auxiliary counting room located in the Administration Building could be utilized. Also, another counting room will be located in the Simulator Building now under construction. The Simulator Building is located near the Visitors Center. Radiation levels will be acceptable for counting equipment and personnel access. No design modifications are necessary.

5. Radwaste Buildings (Unit 3 LOCA) -
A small portion of the Interim Radwaste Building will be inaccessible for approximately the first four days of a LOCA. However, the area of the building containing waste system controls is accessible at all times and has sufficient shielding to meet GDC 19. A review of the planned Radwaste Building south of Unit 3 has shown the building to be accessible. The building control room has sufficient shielding to meet GDC 19. No access problems are anticipated in either building if a LOCA occurs in Units 1 or 2. No design modifications are necessary.

6. Areas of Turbine Building Adjacent to Auxiliary Building -
The location of the Reactor Building personnel hatch "looks" into the Turbine Building across an open area in the Auxiliary Building. Areas of the Turbine Building that "see" the personnel hatch will not be accessible for 4-7 days, depending upon proximity to the Auxiliary Building. The areas of the Turbine Building that are affected are very dependent upon which unit the LOCA occurs in. Regardless of the unit involved, no vital functions are controlled from or centered in the affected areas of the Turbine Building. Access from one end of the station to the other is not prevented. No design modifications are necessary.

7. Low Pressure Injection System Valve Room -
Located in the LPI pump valve area are manual valves that would have to be operated if chemical addition to the Containment sump became necessary. Accessible locations for these valves have been identified. Design drawings will be released by July 1 with installation anticipated to be complete by January 1, 1981.

8. Chemical Addition Area of Auxiliary Building -
Station emergency procedures call for the addition of caustic to Containment sump 30 minutes after initiation of recirculation, if required. Results of final calculations show that the Caustic Addition Tank, the Caustic Addition Pump, and related controls are accessible within 1 hour after a LOCA. No design modifications are necessary.

9. Hydrogen Purge Unit Connections -
The present piping connections for the HPU are located in the Mechanical Penetration Room of each unit. The routing of low pressure recirculation piping in the penetration rooms preclude access to the HPU piping connections. The piping connections for the Hydrogen Recombiner will be modified to accept the HPU. The connections will be accessible within one day after a LOCA. The implementation schedule is identical to that for the Hydrogen Recombiner.

With the issue of IE Bulletin 79-01B, the review of equipment qualification was greatly expanded in scope. As a result, the findings of the radiation qualification review will be submitted with the 79-01B 90 day response which will be available prior to June 1, 1980. Any design modifications required due to insufficient radiation qualification are anticipated to be complete by January 1, 1981.

Accessible areas for post-accident sampling hoods have been located. In addition, some tubing re-routing will be required. A more detailed discussion is in Section 2.1.8.a.

No additional shielding of the RB atmospheric sample line has been identified. This line is not a significant contributor to area radiation levels. However, new monitoring equipment is expected to be located in areas where this sample line is a contributor. At that time, this line will be shielded, as well as other contributors to area radiation, as required to reduce area radiation levels in the vicinity of the new equipment.

2.1.8 Instrumentation to Follow the Course of an Accident

a. Improved Post-Accident Sampling Capability

New sampling points have been selected to allow collection of pressurized and unpressurized reactor coolant samples. Pressurized and unpressurized reactor coolant will be collected from the cold leg drain line on each unit. A sump sample will be collected from the low pressure injection system coolers. The pressurized and unpressurized reactor coolant and sump sample lines will be routed to a sampling hood designed to reduce radiation exposures during sample collection.

In addition to the reactor coolant and sump samples, a containment atmosphere sample line will also be routed to this sampling hood. The containment atmosphere sample will be obtained from the hydrogen analyzer sample lines.

The design of this sampling hood is being finalized for equipment purchase by April 1, 1980, with functional testing in late summer and equipment delivery on site by November, 1980. Installation schedule has not been established.

In the interim, if a nuclear accident should occur and fuel damage above 1% is suspected, procedures for collection of reactor coolant, sump water and containment air samples have been revised to incorporate actions to be taken to minimize radiation exposures during collection and transport of samples to the laboratory(s). These procedures specify the preplanning to be performed as well as notifications and approvals required prior to sample collection. Samples can be collected within one hour in all instances where personnel exposure does not exceed 3 rem/qtr whole body and 18 3/4 rem/qtr to the extremities. In the extremely unlikely event that an accident and extensive fuel damage occurs which results in the predicted personnel exposures exceeding the above, samples can still be collected. However, this cannot be accomplished within one hour. The time required to install additional shielding and allow sample collection and minimize personnel exposure will be dependent upon the nature of the event. Duke considers that these actions are sufficient for the interim time frame until the new sample hood described above is functional.

Analytical procedures have been reviewed and have been determined to be adequate for samples of low and intermediate activity. The probability of an accident which will result in extremely high levels of activity in this interim time frame is considered to be very low. Upon installation of the above described sample hood, sample collection ability under any situation will be available.

b. Increased Range of Radiation Monitors

Procedures have been revised to enable estimation of high level noble gas and radioiodine releases made during accident conditions if the existing effluent instrumentation goes off scale.

The proposed monitors supplement the present process and effluent monitors which are described in Section 11.1.2.4 of the FSAR. This information in conjunction with previous submittals provides the required information on monitor operation and calibration. Procedures for the collection of gaseous effluent samples have been revised to incorporate actions to be taken to minimize radiation exposures during collection and transport of samples to the laboratory(s) if

the vent gas radiation monitors read off-scale. These procedures specify the preplanning steps to be taken as well as the notifications and approvals required prior to sample collection.

Releases through the steam relief valves are not monitored with a gas effluent radiation monitor. The main steam lines from each unit are monitored by coaxial ion chambers which have a range of $0.1\text{-}10^7$ mR/hr. These monitors are located within the Turbine Building, on each main steam line, 1 foot from the Turbine Building wall and about 7 feet from the closest steam relief valve. A procedure is being developed to quantitate releases from the steam relief valves if steam generator tube ruptures exist at the time of a nuclear accident involving fuel damage. This procedure is expected to be fully implemented by March 15, 1980.

c. Improved Inplant Iodine Monitoring

A procedure to determine airborne radioiodine concentrations has been established which does not rely on the availability of a counting room. This procedure utilizes portable "survey-type" instrumentation to determine a "go" or "no go" iodine concentration for respiratory equipment use. This instrumentation in conjunction with silver zeolite cartridges, is a fully adequate method to monitor iodine inplant. This instrumentation is located within the Control Room/Technical Support Center.