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DENTON,H.R. Office of Nuclear Reactor Regulation

SUBJECT: Supplements util 791019,1121 & 1217 responses to NUREG-0578. Included info re emergency power supply requirements, instrumentation for detection of inadequate core cooling & integrity of sys outside containment.

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MAY

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TELEPHONE: AREA 704
373-4083

January 2, 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 Reactor Branch No. 4

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

Dear Sir:

This letter supplements my letters of October 18, November 21, and December 17, 1979 concerning implementation of the recommendations contained in NUREG-0578.

The letters by Mr. A. C. Thies of December 17 and 27, 1979, as well as my letters, provide detailed justification for the proposed implementation schedule of Category A items which will not be completed by January 1, 1980. This justification and proposed schedule remain valid.

It continues to be our intention to complete all of the remaining Category B items prior to January 1, 1981. However, as indicated in the attached, equipment delivery dates may preclude accomplishment of this goal. As final designs are completed and schedule for receipt of equipment established, we will provide our schedule for implementation of the Category B, NUREG-0578 items.

Very truly yours,

William O. Parker, Jr.

William O. Parker, Jr.
Vice President, Steam Production

RLG:scs

Attachment

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DUKE POWER COMPANY

RESPONSE TO NUREG-0578
SHORT-TERM RECOMMENDATIONS
FOR
OCONEE NUCLEAR STATION

Supplemental Response
January 2, 1980

DUKE POWER COMPANY

Response to NUREG-0578
for
Oconee Nuclear Station
Supplement

January 2, 1980

2.1.1 Emergency Power Supply Requirements

Based on the descriptions provided in my letter of November 21, 1979 and the positions and clarifications provided by the NRC Staff, it is concluded that the Oconee design meets the intent of this item.

2.1.2 PWR Relief and Safety Valve Testing

By letter dated December 17, 1979, Mr. W. J. Cahill, Jr., Chairman of the EPRI Safety and Analysis Task Force submitted "Program Plan for the Performance Verification of PWR Safety/Relief Valves and Systems," December 13, 1979.

Duke considers this program to be responsive to the requirements of this item. The EPRI Program Plan provides for a completion of the essential portions of the best program by July 1981. Duke will be participating in this program to provide technical review and to supply plant specific data, as required.

2.1.3.a Direct Indication of Power-Operated Relief Valve and Safety Valve Position

The system description and schedule for installation previously provided by my letter of November 21, 1979 remain valid.

2.1.3.b Instrumentation for Detection of Inadequate Core Cooling

The response provided by my letter of November 21, 1979 remains valid. The procedure for use of existing instrumentation in determining adequacy of core cooling has been implemented, and is available on site for NRC review. By January 31, 1980, the following will be provided as available:

1) A description of additional instrumentation proposed for installation

2) A description of the functional design requirements

3) A description of the procedures to be used and the analysis used to develop these procedures

4) An installation schedule

2.1.4 <u>Containment Isolation Provisions</u>

The design description and schedule for installation previously provided by my letter of November 21, 1979 remain valid.

2.1.5.a <u>Dedicated Penetration for Hydrogen Control</u>

The penetrations to be used for post accident hydrogen control are two inch Reactor Building air sample and return penetrations. They have no use other than post accident hydrogen control and air sampling which is a necessary function for hydrogen control. The only change required to comply with this item is the addition of another remote isolation valve on each penetration inside containment in order to meet the single failure criteria. The addition of the two Reactor Building isolation valves per unit will require shutdown of the unit. The exact schedule for completion of this modification has not been established. The valves themselves will require approximately a year to obtain.

2.1.5.b,c No response is required for Oconee.

2.1.6.a <u>Integrity of Systems Outside Containment Likely to Contain Radioactive Materials</u>

Radioactivity released from systems contained in the Auxiliary Building to tanks and/or the atmosphere, whether liquid or gaseous, will be detected by one of the following two methods which briefly are:

1. <u>Leakage Calculations</u> - are performed that takes into account the entire reactor coolant system, all waste water collection tanks and sumps.

2. <u>Radiation Monitors</u> - will detect the gaseous activity released to the Auxiliary Building atmosphere.

As part of our immediate leak reduction program we intend to incorporate leakage identification and leakage testing programs.

The following actions are performed continuously or, at a minimum, once per day, to identify leakage:

<u>Continuous</u>

(1) The Auxiliary Building atmosphere is sampled by a multipoint gaseous radiation monitor (RIA-32)

(2) The unit ventilation stack is sampled by particulate (RIA-43), iodine (RIA-44) and gaseous (RIA-45) radiation monitors

<u>Every 24 Hours (minimum)</u>

(1) A reactor coolant system leakage test is performed to quantify any such leakage.

(2) A total water inventory program is performed.

(3) Visual Surveillance of accessible areas containing operating systems included in the Leakage Reduction Program is performed.

As stated above, these Leakage Programs are performed at least once per day (every 24 hours) and if an abnormal increase in leakage (more than 2 gpm) or an unexplained increase in the Auxiliary Building/Unit Vent Radiation Monitors is identified, the frequency of the leakage calculations will be increased and the shift personnel perform an evaluation and investigate all possible sources by performing a leakage identification procedure.

The systems which are included in the Leakage Testing and Reduction Program are identified as follows:

1. High Pressure Injection System which encompasses, High Pressure Injection Recirculation, makeup and letdown.
2. Low Pressure Injection System encompasses Residual Heat Removal System
3. Building Spray System encompasses Containment Spray Recirculation
4. Gaseous Waste Disposal System encompasses Waste Gas
5. Purification and Deborating Demineralizers
6. Chemical Addition and Sampling Systems
7. Coolant Storage System
8. Coolant Treatment System

The systems which are excluded from the program are identified as follows:

1. Reactor Coolant System
2. Spent Fuel Cooling System
3. Demineralized Water System
4. Liquid Waste Disposal System
5. Fuel Cask Decon System
6. Auxiliary Service Water System
7. Bulk Hydrogen System
8. High Pressure Service Water System
9. Low Pressure Service Water System
10. Reactor Building Purge and Penetration Room Ventilation System
11. Condensate System
12. Feedwater System
13. Vacuum System
14. Main Steam and Auxiliary Steam System
15. HP and LP Turbine Exhaust and Steam Seal System
16. Moisture Separator and Reheater Heater and Drain System
17. Feedwater Heater Drain and Vent System
18. Recirculated Cooling Water System
19. Treated Water System
20. HP Feedwater Heater Blanket System
21. Nitrogen Purge and Blanket System
22. Startup Steam System

23. Condenser Circulating Water System
24. Turbine Lube Oil Purifying, Fuel Oil and Atomizing System
25. Compressed and Breathing Air Systems
26. Component Cooling System
27. Plant Heating and Condensate Heating System

Boundaries have been defined for these systems and are identified. The boundary is considered to be the total system as shown on the appropriate system flow diagrams. This included all piping, valves, equipment, etc. throughout the system and inclusive of the first isolation valve for the various system interconnections as well as vents and drains. Potential leakage sources are also defined and are identified in the specific periodic leak test procedure.

Fluid systems will be tested under normal operational conditions (temperature and pressure) which it will be exposed to, during normal and abnormal operating conditions. Potential leakage sources will be inspected for leakage and documented as to the amount of leakage found. (e.g., less than 1 ml/min or actual leakage measured). If excessive leakage is identified immediate steps will be performed to: a) isolate the leakage, b) contain the leakage, and c) repair/replace the component.

Gaseous systems will be tested under normal operational conditions (temperature and pressure) which it will be exposed to during normal and abnormal operating conditions. In an attempt to reduce the amount of gas generated, the gas decay tanks will be tested by the following methods:

a) Pressure Decay method to determine their leakage rate, and

b) Soap Bubble method to determine specific component leakage

The remaining portion of the system will be tested by injection of helium into the system until a pre-determined pressure is reached; then the potential leakage sources will be checked for leakage. The leakage rate will be determined by the flow rate of the helium entering the system after the stable pressure condition has been reached.

If leakage is identified, immediate steps will be taken to:

a) identify and isolate the leak

b) contain the leak

c) repair/replace the component

An extensive review of the systems was conducted for the purpose of identifying boundaries and potential leakage sources. During the course of this review we considered IE Circular 79-21 and have concluded that no leak path exists similar to that which occurred at North Anna I.

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

10CFR20 and General Design Criterion (GDC) 19 of Appendix A to 10CFR50 require control of radiation exposure to personnel associated with nuclear station operations. In addition, GDC 4 of Appendix A to 10CFR50 requires safety equipment and systems to function in the environmental conditions to which they either will or may be subjected during the station lifetime. A review of the Oconee Nuclear Station was initiated to determine if any areas of the station failed to meet the above criteria. Personnel access criteria is as recommended in Harold Denton's October 30, 1979 letter to all operating nuclear power plants. These criteria are:

1) Less than 15 mR/hr for areas requiring continuous occupancy and

2) GDC 19 (5 rem whole body or equivalent to any organ) for areas requiring infrequent access.

Equipment suitability criteria is by comparison of calculated environmental conditions with the equipment design and/or qualification.

The accident scenario selected to yield the greatest release of radioactivity from the Reactor Coolant System (RCS) is the Loss of Coolant Accident (LOCA) with subsequent fuel damage. The basis for selecting this particular scenario as the Design Basis Accident (DBA) is discussed in TID-14844. The resulting airborne activity assumed to be released to the containment is 25% core inventory of iodines and 100% core inventory of noble gases. These values are consistent with Regulatory Guide 1.4 and TID-14844. Typically, the liquid activity has been assumed to be 50% core inventory of iodines and 1% core inventory of the remaining fission products. These values are consistent with Regulatory Guide 1.7 and TID-14844. However, Harold Denton's October 30, 1979 letter recommended the inclusion of 100% core inventory of noble gases with the previous liquid activity. Our calculations show that less than 2% of the noble gas inventory will remain in solution post-LOCA. Although we consider the inclusion of noble gases with the liquid activity to be an unnecessary conservatism, we have accepted the NRC Staff recommendation for our initial station review. As a result, the fission product distribution assumed for the initial Oconee Station review is:

Airborne: 100% core inventory of noble gases
25% core inventory of iodines
(These activities are assumed to be homogeneously distributed throughout the containment free volume.)

Liquid: 100% core inventory of noble gases
50% core inventory of iodines
1% core inventory of remaining fission products
(These activities are assumed to be homogeneously distributed throughout a water volume consisting of: RCS, Core Flood Tanks, and water injected by the HPI/LPI Systems.)

To aid in identifying potential personnel access problems, the station was divided into post-LOCA radiation zones. Systems considered in determining the post-LOCA radiation zones were: residual heat removal, recirculation, letdown, and radwaste. Color coded general arrangement drawings showing the radiation zones were provided for review by station operations personnel. The major emphasis of the review was to assure that station personnel would be able to carry out their emergency procedures and maintain all Technical Specification commitments. The following areas have been initially identified as potential locations where personnel exposures may exceed the stated dose rate criteria:

1. Control Room.

2. Administration Building (if LOCA is in Unit 1)

3. Administration Building Annex (if LOCA is in Unit 1)

4. Counting Room (if LOCA is in Unit 2)

5. Radwaste Building (if LOCA is in Unit 3)

6. Areas of Turbine Building adjacent to Auxiliary Building locations of Reactor Building personnel hatch

7. Low Pressure Injection System valve room

8. Chemical Addition area of Auxiliary Building, and

9. Penetration rooms containing connections for Hydrogen Purge Unit.

Design revisions under active consideration for resolution of the above potential areas of concern are: 1) addition of fixed shielding, 2) system redesign, and 3) optimization of post-accident procedures. A schedule of implementation of design revisions will be supplied as soon as designs are finalized.

In addition to determining radiation zones, integrated exposures are being calculated for use in evaluating equipment radiation qualification. A location specific review is currently underway to identify any potential areas of concern. Resolution of potential areas of concern regarding equipment qualification is intimately associated with resolution of potential personnel access concerns. Because of this, review of specific equipment radiation qualification will not be completed until design revisions for personnel access are finalized.

Finally, a review of the present station sampling system was conducted to judge the adequacy of its shielding design. (The sampling system is briefly discussed in this section because it is the only system where field-routed lines are a post-LOCA concern.) Initial results indicate that the sampling system will not meet the dose rate criteria discussed earlier. Design revisions under consideration are: 1) addition of fixed shielding and 2) system redesign.

A schedule for implementation of design revisions will be supplied as soon as design requirements are finalized. A more detailed discussion of the sampling system is found in Section 2.1.8.a.

2.1.7.a Automatic Initiation of the Auxiliary Feedwater System

The system description provided by my letter of November 21, 1979 remains valid.

The following is the proposed method of upgrading the TDEFWP initiation circuits to the safety grade.

The Turbine Driven Emergency Feedwater Pump (TDEFWP) initiation circuits will be upgraded. The pressure switches are safety grade and independent of the Motor Driven Emergency Feedwater Pumps (MDEFWPs). The control switch is also safety grade. However, the power to steam supply valve MS-93 need not be safety grade since a loss of power will start the TDEFWP automatically.

The above changes in conjunction with previously submitted safety grade Motor Driven Feedwater pump initiation circuits will satisfy NUREG-0578 requirements regarding safety grade feedwater initiation.

Pursuant to direction received from the Staff at the regional meeting and in the implementation schedule table provided in H. R. Denton's letter of October 30, 1979, the above is provided for Staff review and approval prior to implementation. Implementation of this modification at Oconee prior to January 1, 1981 is dependent upon Staff approval in a timely manner.

2.1.7.b Auxiliary Feedwater Flow Indication

The schedule for upgrading the present flow indication system to safety grade previously provided by my letters of November 21, 27, 1979 remain valid.

2.1.8.a Improved Post-Accident Sampling Capability

A design and operational review of the reactor coolant and containment atmosphere sampling systems has been performed for post-accident sampling. As anticipated, current sampling locations do not provide an accurate representation of reactor coolant conditions during an accident since pressurized reactor coolant is sampled from the pressurizer and unpressurized reactor coolant from the letdown. Primary sample taps will be relocated to assure a representative sample of pressurized reactor coolant under conditions of natural circulation. Relocation of these taps will be coordinated with scheduled outages and equipment availability in an attempt to have these installed by January 1, 1981. Unpressurized reactor coolant samples will be obtained from the LPI system.

Sample hoods for pressurized and unpressurized reactor coolant must be relocated to reduce radiation levels in the auxiliary building from long pipe runs, and be redesigned to limit radiation exposures during sample collection. Sample hood locations have been selected but are under review to determine accessibility under accident conditions. Hood locations and hood design information will be made available as soon as possible in an attempt to have these installed by January 1, 1981.

At this time in our review, shielding will need to be added to sampling lines so that occupational exposures can be reduced while obtaining reactor containment atmosphere samples.

Existing procedures provide for prompt radiological spectrum analyses of noble gases, radioiodines, radiocesiums, and other nonvolatile radionuclides. No difficulties are expected in performing these analyses provided samples are promptly prepared in the sample area and the site is accessible since there is a primary and a secondary counting room on site.

The boron and chloride analysis procedures appear adequate in their present form for highly radioactive samples and are capable of being completed promptly (boron analysis within an hour and chloride analysis within eight hours). However, we question the need for these analyses immediately since routine sampling of the borated water storage tank should insure that adequate boron is present in the reactor coolant system after an accident to provide a safe shutdown margin and, since corrosion considerations are a long term concern, also question the need for immediate chloride analyses.

2.1.8.b <u>Increased Range of Radiation Monitors</u>

<u>Unit Vent Monitors</u>

Unit vent monitors for noble gases will be provided for each unit with a range adequate to cover normal and accident conditions. Three monitors will be required to measure activities from 1×10^{-7} μCI/cc to 1×10^{5} μCi/cc of noble gases. These monitors will have at least one decade of overlap.

Continuous indication of unit vent radiation level and the appropriate alarms will be provided in the Control Room.

Primary and secondary calibrations shall be performed in the following manner:

A primary calibration shall be performed on one of each type of monitors in this specification. The primary calibrations shall be accomplished using three (3) National Bureau of Standards (NBS) certified radioactive sources of a high, medium, and low MeV energy yield. The primary calibrations shall be performed at a minimum of two (2) levels of activity. Accuracy, countrate response to energy, range, background response, and minimum detectable concentration shall be determined.

At the time of the primary calibration, the secondary calibration source shall be established.

This secondary calibration source is described as the source that when placed in a repeatable geometry (fixed by a hole, cup, or device) shall check the gain, sensitivity and detector calibration integrity. This secondary calibration source shall be the long half-life source used in conjunction with the primary calibration sources. This secondary source shall be traceable to documentation to primary calibration sources traceable to the National Bureau of Standards.

The calibration will be performed annually as required by Technical Specifications.

The equipment will be purchased by March 1, 1980, but because of long lead time on vent monitors, delivery will be by March 1, 1981 with installation by May 1, 1981.

Containment Radiation Monitors

Two physically and electrically separated containment radiation monitors shall be provided to monitor 10^8 Rad/hr. These monitors shall be qualified to IEEE-323, 1971 and powered from the vital instrument buses.

The monitor output shall be indicated continuously in the Control Room. These monitors shall be calibrated as indicated above for the vent monitors at refueling as required by the Technical Specifications.

Procedures have been developed to quantitate releases from the unit vents, waste gas decay tanks, main condenser air ejector and auxiliary building and are available on-site for review.

The monitors will be purchased by March 1, 1980 with delivery prior to January 1, 1981. However, monitor installation will require unit shutdown.

Pursuant to direction received from the Staff at the regional meeting and in the implementation schedule table of H. R. Denton's letter of October 30, 1979, the above is provided for Staff review and approval prior to implementation.

2.1.8.c Improved Inplant Iodine Instrumentation

Based on the description provided in my letter of November 21, 1979, and the positions and clarifications provided by the NRC Staff, it is concluded that Oconee presently meets the requirements of this item.

2.1.9 Transient and Accident Analysis

By letter dated December 13, 1979, J. H. Taylor, Manager, Licensing, Babcock and Wilcox, provided "B & W LOFT L31 Pretest Prediction Report" to the NRC Staff.

The program described in my letter of November 21, 1979 and during previous meetings between the B & W 177 FA Owners Group and the Staff is in progress. Initial draft Safety Sequence Diagrams have been received and are undergoing extensive internal review and comment. It currently appears that operating guidelines will be available in August, 1980 with final procedures written and operators trained within six months thereafter.

Containment Pressure Indication

Two identical safety class pressure transmitters will monitor the Reactor Building (RB) pressure and provide signals to Control Room indicators, (one per transmitter), and a shared chart recorder. Each channel will be powered by vital instrument busses. Each transmitter will be located outside the RB and will monitor the pressure with a bellows sensor coupled with a filled capillary tube. Each transmitter will have its own separate independent containment penetration and will be completely independent from the other channel. This instrumentation will meet Regulatory Guide 1.97, dated December 1975.

Each transmitter will monitor a range of 5 psig to 175 psig, a range of three times the RB design pressure.

Containment Water Level Indication

The Reactor Building (RB) water level will be monitored by a wide range and a narrow range system. The narrow range level transmitter will be qualified to Regulatory Guide 1.89, dated November 1974. The transmitter shall be powered from the vital instrument busses and will provide Control Room indication and will be monitored by the plant computer. A bellows sensor coupled to the transmitter mounted at the lowest possible level in the containment primary sump. This transmitter shall have a range of 0 - 3' (one foot above the containment floor.)

The wide range level monitors shall be qualified to meet Regulatory Guide 1.97, dated December 1975.

The independent wide range level transmitters shall monitor the level from the containment floor to a level of 15' or 600,000 gallons. Each transmitter shall provide a Control Room indication with an input to a shared chart recorder. Each transmitter shall be powered from the vital instrument busses. A bellows sensor coupled by a filled capillary tube to the transmitter shall be seismically mounted as close to the floor as possible.

Containment Hydrogen Indication

The primary function of this instrumentation is to provide the operator with information regarding the Reactor Building (RB) atmosphere hydrogen concentration level during and following a LOCA. The instrument shall remain functional during a LOCA in order for the necessary transient and level hydrogen concentrations to be monitored and recorded for future accident analysis. Secondary functions include service testing of the recombiner, providing the operator with information regarding proper functioning following an accident, or whether corrective action (replacing the catalyst bed or unit, initiating containment purge) is required.

Two separate identidal analyzer systems will be installed per unit. These analyzers operate independent of the recombiner system and will be vital sources of power. Each analyzer will be able to monitor either of two identical containment sampling headers or the calibration gases. Each analyzer shall have, along with control panel indicator and alarm, a separate Control Room indicator and alarm with a shared chart recorder.

Each containment sample header will have five inlet samples available for monitoring:

1. Top of containment

2. Operating level

3. Basement

4. Radiation Monitor/Recombiner Inlet header

5. Radiation Monitor/Recombiner Discharge header

All sample selection and switching is accomplished manually by the operator from the remote analyzer control panel. Each analyzer shall have its own sample and return containment penetrations.

Reactor Coolant System Venting

The high point venting system of the reactor coolant system will consist of four solenoid operated vent valves, two in series on the existing high point vent lines for each of the two reactor hot legs.

The existing one inch high point vent lines of the reactor coolant system will be used by adding a tee in each of these lines to tie in the new remotely operated vent valves. The new tee and valves for each of these vents will be outside the secondary shield walls to minimize the effect of these new valves on the reactor coolant system.

Each of the new valves will be 1/2 inch stainless steel solenoid operated valves. These valves will be operated from the unit's Control Room. The valves will have to be energized to open. It will be necessary to have two buttons depressed simultaneously to accomplish venting. Each valve will have remote position indication. Power will be available from a safety grade source capable of being supplied from an emergency bus. However, the power will be removed from the valves to assure that they are not accidently opened during operation.

Each of the two new vents will be discharged into the air being discharged fron the RB Emergency Cooler ducts. The two discharge points are on opposite sides of the RB to provide mixing to minimize the buildup of hydrogen concentration within the containment. Each of the two vents are individually capable of venting one-half the reactor coolant system volume per hour.

There are a number of different ways being investigated to determine the most reliable method of determining when to vent and when to stop the vent. We have not selected a method at this time. We are investigating mechanical/electrical means as well as venting periodically for a short time internal based on the maximum expected gas generation rate expected.

In addition to the two new high point reactor coolant vents, the gases that accumulate in the pressurizer can be vented by use of the PORV presently installed on the pressurizer. We do not propose to add a remotely operated vent to the reactor head, since any accumulation of gases sufficient to fill the reactor vessel volume will be vented via the hot leg vents due to the free path available.

2.2.1.a <u>Shift Supervisor Responsibilities</u>

The Shift Supervisor's responsibilities, as contained in the Steam Production Administrative Policy Manual, has been redefined to include several concerns of this item. With this change, and the existing Technical Specifications and station directives, it is considered that the intent of this item is fulfilled. All of these documents are available on site for NRC review.

2.2.1.b <u>Shift Technical Advisor (STA)</u>

The two functions of the STA, accident assessment and operating experience assessment, have been fulfilled in manner described in my letter of November 21, 1979.

2.2.1.c <u>Shift and Relief Turnover Procedures</u>

Station Directive 3.1.8, "Shift Relief and Turnover," has been revised to fulfill the requirements of this item. This document is available on site for NRC review.

2.2.2.a <u>Control Room Access</u>

Station Directive 3.1.31, "Control Room Access and Authority," has been issued to fulfill the requirements of this item. This document is available on site for NRC review.

2.2.2.b <u>Onsite Technical Support Center (TSC)</u>

The November 21, 1979 letter described the concept of a Technical Support Center (TSC) as is presently utilized in the current Oconee Emergency Plan. By January 1, 1981 this will be revised to describe the use of the two TSC's for both data evaluation and management decision making. To implement this change, a base station connection will be provided in each TSC to allow communication by radio to the local Civil Defense Emergency Operations Centers and to radiological monitoring teams doing environmental surveillance.

Other changes needed to fully implement the requirements for a TSC are administrative and procedural in nature and will be completed by January 1, 1981.

2.2.2.c Onsite Operational Support Center

The Operational Support Center (OSC) has been implemented. Station Directive 3.1.32, "Operational Support Center," provides a description of the OSC and is available on site for NRC review.

A definition of the present OSC was given in our November 21, 1979 letter. As noted in 2.2.2.b above, the definition of the OSC will be changed prior to January 1, 1981 to recognize the TSC as the management decision making point. This change may also require a change in the location of the OSC.