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DENTON,H.R. Office of Nuclear Reactor Regulation
REID,R.W. Operating Reactors Branch 4

SUBJECT: Supplements 791018 ltr re implementation of NUREG-0578 recommendations.Intends to implement each item as required. Power operated relief valve & valve position indication installations require access to containment.

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WILLIAM O. PARKER, JR.
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STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

November 21, 1979

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:

This letter supplements my letter of October 18, 1979 concerning implementation of the recommendations contained in NUREG-0578 as augmented by the Staff letters of September 13, and October 30, 1979, the regional and topical meetings held with the industry during September and October, and a telephone discussion held with the Staff on November 9, 1979. Attached please find our responses to the various items contained in NUREG-0578.

With regard to the implementation schedule, it continues to be our intention to proceed as indicated in our responses with our best efforts towards implementation of each item as required. In this regard, the following specific points are pertinent for the items both identified by the NRC as requiring completion by January 1, 1980 and which require unit shutdowns to complete installation.

1. The PORV and relief valve position indication installations require access to containment and work in the vicinity of the pressurizer. It is required, therefore that the unit be in cold shutdown for installation of the sensors as well as installation of equipment and cabling inside containment. It is estimated that installation and checkout of the indication system would take one week inside containment. The remainder of the work, outside containment, could be accomplished during any plant condition in about two weeks. The required equipment is onsite and the final design is complete.

2. The installation of a diverse containment isolation actuation signal similarly requires a unit outage to complete. The modification requires work in energized Engineered Safeguards cabinets, on actuators of containment isolation valves, and testing and checkout following completion of the modification. The design work for this modification is not expected to be complete until mid-December for Unit 1, and Units 2 and 3 at later dates, therefore the actual manpower and installation requirements have not been

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determined. It is expected, however, that this modification would take a three week unit outage to complete.

In order to minimize unit downtime, it is highly desirable to complete both of the above modifications concurrently during a single outage. Available manpower during the next several months dictate that the modifications be installed on each unit sequentially. Oconee 1 will be shutdown for refueling on November 21, 1979 and is expected to be shutdown for eleven weeks. A total of some 3,000 separate work activities is scheduled for completion during this outage. This includes ten separate NRC commitments, and among these commitments, the fire protection modifications, high pressure injection cross connection, and work associated with IE Bulletins 79-02 and -14 require particularly large manpower resources. A breakdown of available personnel who will be directly involved in the outage is provided below:

Organization	Number of Persons
Oconee Nuclear Station	145
General Office Engineering and Maintenance Support	410
Construction Department	35
Vendor	155

As an example of the manpower requirements, 160 Oconee, General Office, and vendor personnel will be committed to the work required by IE Bulletins 79-02 and -14. Without delaying the above efforts and subsequently extending the refueling outage, there is no extra manpower available to accomplish these two NUREG-0578 modifications on Oconee 2 and Oconee 3 prior to January 1, 1980.

Extending the Oconee 1 outage is not desirable due to system generation requirements as well as the fact that Oconee 2 requires refueling and is intended to be shutdown in the February-March 1980 time frame. The Duke Power system generation requirements for the winter of 1979-1980 are as follows:

System rated capacity (approximate)		12,500 mw
Expected winter peak	10,002 mw	
Average forced outage (1976-1978)	1,134	
Oconee 1	860	
Other scheduled outages	94	
	12,090 mw	
Expected peak excess capacity		410 mw

Further reduction of on peak excess capacity may be expected due to extreme cold weather (up to 950 mw), additional peak forced outages, and adverse weather conditions such as frozen coal, icing problems, and river flow considerations. These winter conditions could occur in the January through late February time frame in our service area. As can be seen, removing another Oconee unit from service

during this period would result in an expected peak deficit of 450 mw. Purchase power from neighboring utilities may or may not be available to offset power deficits.

In consideration of the above, it is our current intent to complete the two NUREG-0578 modifications, described above, on Oconee 1 during the current refueling outage; on Oconee 2 during the forthcoming refueling outage to be commenced following the restart of Oconee 1 in the February-March 1980 time frame; and on Oconee 3 during an outage currently planned to follow the restart of Oconee 2 or by May 31, 1980 at the latest. As previously stated in my letter of October 18, 1979, all other NUREG-0578 items required to be completed by January 1, 1980 will be completed by this date. It is considered that the delay associated with implementation of these other two NUREG-0578 items beyond the NRC date of January 1, 1980 would have minimal effect on the health and safety of the public, particularly when considering all the other improvements that have been made to Oconee in recent months. Conversely, forced outages of all three Oconee units would cause a severe generation deficit to our service area and in the region.

Very truly yours,

William O. Parker, Jr.

RLG:scs
Attachment

DUKE POWER COMPANY

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

Supplemental Response
November 21, 1979

DUKE POWER COMPANY

Response to NUREG-0578
Short Term Recommendations
for
Oconee Nuclear Station

2.1.1 Emergency Power Supply Requirements

- Pressurizer Heaters

The pressurizer heaters for each Oconee unit are supplied from four non-safety-related motor control centers (MCCs). These MCCs are in turn powered via load centers from the 4160-volt Engineered Safeguard (ESG) buses. The MCCs and their associated load centers are divided among the three 4160-volt ESG buses such that the loss of one entire 4160-volt switchgear will not preclude the capability to supply sufficient pressurizer heaters to maintain natural circulation under hot standby conditions.

The MCCs and their associated load centers are not automatically shed from the 4160-volt ESG buses under either blackout or LOCA conditions. Therefore, power is available to the heaters from either the offsite power system or the onsite emergency power system and the transferring of heater loads or reapplying of heater loads is not required. The emergency power sources available to each Oconee unit including their capacity and capability are described in Section 8.2.3 of the Oconee FSAR.

From the description provided above and considering the clarifications provided by the NRC Staff, it is concluded that the Oconee design meets the intent of the NUREG-0578 requirements for pressurizer heaters.

- Power-Operated Relief and Block Valves

The pressurizer power-operated relief valve (PORV) in each Oconee unit is a DC solenoid-operated pilot valve. The PORV block valve in each Oconee unit is an AC motor-operated valve. The power supplies for the PORV's and their associated block valves are therefore independent and diverse. Power is available to the PORV's solenoids from the 125-volt DC instrument and control battery power system. Battery chargers are provided and are powered from safety-grade MCC's which are capable of being powered from both the offsite power system and the onsite emergency power system. Power is available to the block valves through non-load-shed load centers which are capable of being powered from both the offsite power system and the onsite emergency power system. No manual transfer of motive or control power for these valves is required. These power systems are further described in Section 8 of the Oconee FSAR.

2.1.1 Continued

From the description provided above and considering the NUREG clarifications provided by the NRC staff, it is concluded that the Oconee design meets the intent of NUREG-0578 requirements for PORV's and PORV block valve.

- Pressurizer Level Indication

Each Oconee unit has three channels of pressurizer level indication. These pressurizer level channels are a part of the integrated control system. Their power is supplied from a 125-VDC instrument and control battery through a static inverter. A manual transfer switch is installed to provide back-up power from a regulated AC power source. These power systems are further described in Section 8 of the Oconee FSAR. Procedures are available to assure transfer to the back-up power source in a timely manner.

2.1.2 Performance Testing for BWR and PWR Relief and Safety Valves

Duke Power Company is supporting the industry-wide efforts associated with the testing of PWR relief and safety valves. A program description and schedule is expected to be available for submittal by January 1, 1980.

The tests are currently planned to be completed by July 1981.

2.1.3 <u>Information to Aid Operators in Accidents Diagnosis and Control</u>

a. Direct Indication of Power-Operated Relief Valve and Safety Valve Position for PWR's and BWR's

Direct indication of PORV and safety valve position will be provided for each Oconee unit. Valve position will be monitored by a reliable, single channel system powered from a battery-backed vital bus.

The valve position indication components will be seismicly qualified consistent with the component or system to which they are attached. The valve position indication components will also be environmentally qualified as appropriate for conditions applicable to their location.

Valve position will be indicated and alarmed in the Control Room. Installation of a portion of this modification requires access to containment. Additionally, because the indicator is attached to piping subjected to a high temperature environment and is in the vicinity of the pressurizer, this modification can only be installed while the unit is in cold shutdown. The materials are onsite and the necessary design work will be completed by December 1979. Therefore, this modification will be installed at the forthcoming refueling outages of Oconee 1 and 2 and during a scheduled outage of Oconee 3 prior to May 31, 1980.

Alternate means of determining valve position are available to the operator. These include quench tank level, temperature and pressure, and relief valve discharge piping temperature.

b. Instrumentation for Detection of Inadequate Core Cooling for PWR's and BWR's

Duke Power Company is participating with other utilities with B&W nuclear steam supply systems to provide a generic response to this item. The evaluation which B&W is performing covers all cases of inadequate core cooling and will provide recommendations for additional instrumentation in late January 1980. The scope and schedule of this analysis program has been discussed with the Staff.

By January 1, 1980, procedures for use of existing instrumentation in determining adequacy of core cooling will be implemented. Also, as a result of the generic study of inadequate cooling, new instrumentation requirements will be evaluated.

The plant computer performs the calculations for the subcooling monitor. Inputs from each loop consists of one wide range pressure signal (0-2500 psig from buffered output of ES cabinets) and will consist of two hot leg wide range temperature signals (50-650°F from the NNI cabinets.) The computer uses the wide range pressure inputs to determine each loop's

2.1.3 Continued

saturation temperature. The saturation temperature is compared to each loop's highest hot leg temperature and and an alarm is actuated if margin to saturation is less than 35°F for reactor power greater than 2% or if margin to saturation is less than 50°F for reactor power less than 2%. Two meters continuously display margin to saturation temperature of their respective loops. This will be completed by January 1, 1980.

Attachment one gives the availability data concerning the plant computer. These figures are conservative since they are based on continuous required operation of the computers, and much of the outage time includes preventative maintenance and scheduled enhancements which are performed while the plant is off line.

Attachment two gives the required information on the subcooling meter. The requested data on uncertainty will be provided by January 1, 1979.

The requested back-up capability was previously described in a letter dated April 10, 1979, in response to IE Bulletin 79-05, -05A, Item 3. The following procedures were revised to include actions to cope with primary coolant system voids:

EP/O/A/1800/08	Steam Supply System Rupture
EP/O/A/1800/04	Loss of Reactor Coolant

A graph of the properties of water and saturated steam has been added to the above procedures. Temperature and pressure indications are readily available to the operators. Also, a computer program is available to allow the operator to read selected incore thermocouples for core temperature. Operating personnel have been instructed in these procedural changes.

ATTACHMENT 1

	AVAILABILITY	MINUTES DOWNTIME
1974		
Oconee No. 1	97.57%	13,459
Oconee No. 2	99.17%	4,417
Oconee No. 3	98.87%	6,032
1975		
Oconee No. 1	99.08%	4,805
Oconee No. 2	98.92%	5,640
Oconee No. 3	97.96%	10,388
1976		
Oconee No. 1	98.02%	10,421
Oconee No. 2	97.16%	14,890
Oconee No. 3	99.07%	4,853
1977		
Oconee No. 1	99.21%	4,141
Oconee No. 2	98.79%	6,342
Oconee No. 3	98.48%	7,967
1978		
Oconee No. 1	97.68%	12,160
Oconee No. 2	97.74%	11,846
Oconee No. 3	99.19%	4,246
1979		
Oconee No. 1	99.69%	808
Oconee No. 2	99.31%	1,790
Oconee No. 3	97.20%	7,298

CALCULATOR

Type (process computer, dedicated digital or analog calc.)	Process Computer
If process computer is used specify availability. (% of time)	See Attachment 1
Single or Redundant calculators	Single
Selection Logic (highest T., lowest press)	See Description
Qualifications (seismic, environmental, IEEE323)	N/A
Calculational Technique (Steam Tables, Functional Fit, ranges)	Steam Tables

INPUT

Temperature (RTD's or T/C's)	*Rosemount RTD Model #177GY
Temperature (number of sensors and locations)	2 Hot leg/loop
Range of temperature sensors	50-650^{o}F
Uncertainty of temperature sensors (^{o}F at 1)	To be supplied later
Qualifications (seismic, environmental, IEEE323)	Seismic
Pressure (specify instrument used)	Motorola-Model 56PH
Pressure (number of sensors and locations)	1/loop in Hot Leg
Range of Pressure sensors	0-2500 psig
Uncertainty of pressure sensors (PSI at 1)	To be supplied later
Qualifications (seismic, environmental, IEEE323)	Seismic & environmenta

BACKUP CAPABILITY

Availability of Temp & Press	
Availability of Steam Tables etc.	
Training of operators	
Procedures	

* This model number is for a direct immersion type RTD. These are being replaced on an as needed basis with well type RTD's. (Model #177 HW)

INFORMATION REQUIRED ON THE SUBCOOLING METER

DISPLAY - METER

Information Displayed (T-Tsat, Tsat, Press, etc.)	T-Tsat
Display Type (Analog, Digital, CRT)	Analog
Continuous or on Demand	Continuous
Single or Redundant Display	Redundant
Location of Display	Control Room
Alarms (include setpoints)	N/A
Overall uncertainty (^{o}F, PSI)	To be supplied later
Range of Display	0-999
Qualifications (seismic, environmental, IEEE323)	NA

DISPLAY - COMPUTER

Information Displayed (T-Tsat, Tsat, Press, etc.)	T-Tsat
Display Type (Analog, Digital, CRT)	CRT/Alarm Typer (Poin
Continuous or on Demand	Continuous
Single or Redundant Display	Single
Location of Display	Alarm CRT/Control Room
Alarms (include setpoints)	See Description
Overall uncertainty (^{o}F, PSI)	To be supplied later
Range of Display	N/A
Qualifications (seismic, environmental, IEEE323)	N/A

2.1.4 Containment Isolation Provisions for PWR's and BWR's

Containment isolation valves on systems which are determined to be non-essential will be isolated on a signal diverse from the currently utilized containment pressure signal. Design work is in progress to provide for isolation of non-essential systems at the low RCS pressure trip setpoint.

Those systems considered to be non-essential and which will be isolated on either high Reactor Building pressure or low Reactor Coolant System pressure include:

- Quench Tank sample
- Quench Tank gaseous vent
- Reactor Building purge
- Reactor Building sump drain
- Reactor Building atmosphere sample
- Pressurizer sample
- OTSG sample
- OTSG drain

Those systems considered to be essential include:

- Reactor Coolant Pump Seal Return
- Component Cooling to Reactor Coolant Pumps
- Low Pressure Service Water to the Reactor Coolant Pumps

Containment isolation valve control circuits have been reviewed to identify those isolation valves that could reposition automatically upon reset of the containment isolation (CI) signal. The necessary modifications will be made to assure that resetting the CI signal will not automatically reposition any containment isolation valve. Reopening any valves that are closed by the containment isolation signal will require deliberate operator action. This will be corrected concurrently with providing diverse containment isolation.

Completion of this modification requires extensive work in Engineered Safeguards cabinets as well as to several containment isolation valves. It has been determined that this modification must be accomplished at cold shutdown because of the work in energized cabinets and on isolation valves. The design work will not be completed until late December 1979 for Oconee 1. Two to three weeks will be required to complete the design work for each of the other two Oconee units. It is estimated that a minimum of a three week outage will be required.

This effort will be completed during the forthcoming refueling outages of Oconee 1 and 2 and during a scheduled outage of Oconee 3 prior to May 31, 1980.

2.1.5 Post-Accident Hydrogen Control System for PWR and BWR Containments

a. Dedicated Penetrations for External Recombiner or Post-Accident External Purge System

The post-accident hydrogen control system which is to be installed at Oconee will utilize existing penetrations currently dedicated to reactor building atmospheric monitoring. The piping will be arranged to allow simultaneous monitoring and hydrogen control through one penetration pair. Redundant isolation valves will be added inside each penetration. A detailed description of this modification will be provided by January 1, 1980.

b. Inerting BWR Containments

Not Applicable to Oconee

c. Capability to Install Hydrogen Recombiner at Each Light Water Nuclear Power Plant

Although this item is the subject of NRC rulemaking, Duke is pursuing a design to provide the capability to install a hydrogen recombiner in the existing hydrogen purge lines.

2.1.6 <u>Post-Accident Control of Radiation in Systems Outside Containment of PWR's and BWR's</u>

a. Integrity of Systems Outside Containment Likely to Contain Radioactive Materials (Engineered Safety Systems and Auxiliary Systems)

Periodic leak detection programs and preventive maintenance programs have been in effect at Oconee.

A Reactor Coolant System leakage test each shift is run to quantify any such leakage. This leakage calculation takes into account the entire RCS including those portions outside the containment. If noticeable change is encountered, an evaluation and investigation of possible sources is initiated. Also, personnel monitor total water inventory including levels from all waste water collection tanks and sumps. If unusual increases occur, operating shift personnel are notified and they perform a Leakage Identification Procedure. If a significant leak develops in the Auxiliary Building, gaseous activity released will be detected by the multipoint radiation monitor (RIA-32). The ventilation stack monitors may also identify such leakage (RIA-43 (particulate), RIA-44 (iodine), RIA-45 (gas)).

Prior to each refueling outage, the entire LPI system is inspected for leaks. All piping, valves, and components are visually checked for leakage and a listing is made of areas where repair is required. These repairs which reduce system leakage are completed during the outage prior to restart of the unit. Similar inspections are conducted on the high pressure injection and letdown systems. During periodic performance tests, a similar inspection is made of the Reactor Building spray system.

Additionally, a leakage test is run annually on the entire LPI system. All leaks are carefully measured and recorded. The acceptance criteria for this test is less than 2 gallons per hour.

To augment the leakage detection and formalize the leakage prevention program, the following actions will be taken:

(a) A review of those systems potentially containing significant levels of radioactive fluids in post-accident conditions will be carried out to establish boundaries and identify potential leakage sources.

(b) Administrative action will be taken to establish a requirement for shift personnel to perform periodic visual surveillance of accessible areas containing operating systems defined in (a) to identify and correct leaks.

Both (a) and (b) will be implemented by January 1, 1980.

(c) A leakage testing program will be developed to identify leakage in those systems designated above. Such leakage test will be carried out at each refueling outage (nominally 18 months). Initial leak tests will be performed at the first refueling outage commencing after January 1, 1980 for each Oconee unit.

b. Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems which may be used in Post-Accident Operations

A review is in progress to determine post-accident radiation levels throughout the station. Areas and equipment that are vital and require access will be determined. This review will be completed by January 1, 1980. Based on this review, appropriate corrective actions, including procedure revision, system redesign, or shielding modifications, will be taken to assure accessibility and operability during post-accident radiation conditions. A schedule for the implementation of these corrective actions will be provided January 1, 1980.

2.1.7 Improved Auxiliary Feedwater System Reliability for PWR's

a. Automatic Initiation of the Auxiliary Feedwater System

The emergency feedwater system at Oconee presently has the automatic/manual initiation capability installed. Upon loss of both main feedwater pumps (MFWP), both motor driven emergency feedwater pumps (MDEFWP) and the turbine driven emergency feedwater pump (TDEFWP) start. Each MDEFWP supplies flow to one steam generator through a safety-related flow control valve. The TDEFWP supplies flow to both steam generators through both safety-related flow control valves. Loss of MFWP's is sensed independently for each pump. Each MDEFWP is automatically initiated from low MFWP discharge pressure switches or low MFWP control oil pressure switches. The TDEFWP is automatically initiated from low MFWP discharge pressure switches or MFWP steam supply valve limit switches. Thus, the initiating circuits satisfy the single failure criteria requirements. The MDEFWP's and their initiating circuits are supplied power from the emergency bus. The TDEFWP initiation circuits are battery backed and on loss of power to the initiation circuits, the TDEFWP will start automatically. The initiating circuits will be tested every 18 months during functional system testing. A failure of the automatic initiating circuits will not prevent manual initiation of the AFWS from the control room. Instrument air is supplied to the flow control valves with manual capability to supply nitrogen. In addition, the instrument air compressors are capable of being supplied power from the emergency power system.

The TDEFWP initiation is presently control grade and will be upgraded by the installation of qualified switches, cabling, and other components as required. The MDEFWP's presently have safety grade initiation and no further upgrade is required.

b. Auxiliary Feedwater Flow Indication to Steam Generators

The emergency feedwater system at Oconee presently has flow indication to the steam generators. One control grade transmitter is installed in the flow path (just upstream of the flow control valves) to each steam generator. In addition, control grade level transmitters are installed on each steam generator. Thus, the single failure criteria is satisfied by one control grade flow transmitter and one control grade level transmitter per steam generator. The flow transmitters and level transmitters will be tested during the functional tests of the emergency feedwater system. Power supplies to all the above transmitters are battery backed. The accuracy of the feedwater flow transmitters is within the required $\pm$ 10%.

The flow indication of this system is presently control grade and will be upgraded to safety grade by January 1, 1980.

2.1.8 Instrumentation to Follow the Course of an Accident

a. Improved Post-Accident Sampling Capability

As stated in our response to Item 2.1.6.b, a review is in progress to determine post-accident radiation levels throughout the station, including sample areas. This review should be completed by December 15, 1979. Once this review is completed, new sampling locations will be selected to allow collection of pressurized and unpressirized reactor coolant samples. Current sampling locations might 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. Selection of sampling locations, relocation of sampling hoods to reduce radiation levels in the auxiliary building, and conceptual design of these hoods to limit radiation exposures during sample collection cannot be completed prior to January 1, 1980, due to manpower limitations. These will be provided as soon as they are available.

At this time in our review, we do not anticipate any problems in obtaining reactor containment atmosphere samples and providing a description of any corrective actions required by January 1, 1980.

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, immediate chloride analyses are apparently not needed.

b. Increased Range of Radiation Monitors

A vent monitor for noble gases will be provided with a range adequate to cover normal and anticipated conditions. Multiple monitors will be required to cover this range; one decade overlap will be provided.

Redundant containment radiation monitors will be provided to monitor 10^8 R/hr. The installation will be completed by January 1, 1981. In the interim, procedures will be developed to quantitate releases from the unit vents, waste gas decay tanks, main condenser air ejector, and auxiliary building. Radioiodine releases may be quantitated using silver zeolite cartridges instead of charcoal cartridges to reduce noble gas interference. These procedures will be in place and personnel trained in their use by January 1, 1980.

c. Improved In-Plant Iodine Instrumentation

Silver Zeolite radioiodine sampling cartridges are in use at Oconee for sampling air when the presence of noble gases is suspected. Oconee Health Physics personnel are knowledgeable in the appropriate station procedures required and are trained in the equipment required to determine airborne iodine concentration in the plant under all conditions. It is considered that Oconee is presently meeting the requirements of this item.

2.1.9 <u>Analysis of Design and Off-Normal Transients and Accidents</u>

Duke Power Company, in conjunction with the other B&W 177 FA Owners, is supporting the Abnormal Transient Operating Guidelines Program to address this concern. This program will utilize plant specific system information and available analytical data to investigate realistically a wide range of reactor plant transients, including failures not normally considered in licensing documentation. The result will be the development of appropriate guidelines to enable plant operators to deal effectively with abnormal transients, as well as the promotion of better operator understanding of system fundamentals and abnormal transient operation. It is anticipated that the operating guidelines resulting from this program will be available by mid-1980.

The LOFT L3-1 pre-test prediction report is scheduled for submittal January 15, 1980. Babcock & Wilcox, the Oconee NSSS vendor, is developing computer code model information prior to the test, which will be retained by the NRC staff.

2.2.1 Improved Reactor Operations Command Function

a. Shift Supervisor Responsibilities

Appropriate directives will be reviewed and revised as necessary to fulfill the intent of this item by January 1, 1980.

b. Shift Technical Advisor (STA)

The two functions of the STA, namely accident assessment and operating experience assessment, will be fulfilled in the following manner.

An experienced SRO who has been instructed in additional academic subjects will be provided on each shift by January 1, 1980. It is intended that he will provide the on-shift accident assessment capability. Further training will be conducted through 1980 to meet the intent of this item. These SRO's will be detached from and independent of the normal line function of plant operation. He will be an advisor to the Shift Supervisor.

For the second function, operating experience assessment, several engineers will be assigned. It is anticipated that they will be familiar with plant operations, represent diverse technical backgrounds and be supplemented with additional training in operations. These engineers will report to station management other than shift personnel. These assignments will be made before January 1, 1980.

c. Shift and Relief Turnover Procedures

Station Directive, "Shift Relief and Turnover," will be revised to fulfill the requirements of this item by January 1, 1980.

2.2.2 Improved In-Plant Emergency Procedures and Preparations

a. Control Room Access

Appropriate procedures and directives will be reviewed and revised as necessary by January 1, 1980, to meet the concern of this item.

b. Onsite Technical Support Center

The Oconee Nuclear Station has established an onsite Technical Support Center (TSC). This TSC is located on the north side of the Control Room area for Units 1 and 2 and Unit 3. An Operations Center (OC) is established in the Administration Building conference room area. A near-site Crisis Management Center (CMC) is established in the Visitors Center basement area.

Individuals staffing the TSC are the Duty Operating Engineer, Duty Health Physicist, Chemist, I&E Supervisor, Maintenance Engineer, Performance Engineer and technicians necessary to support them. The OC is staffed by the Station Manager and the Superintendent of Operations, Technical Services, Maintenance and Administration and the Station Health Physicist. These centers are manned when the Station Manager determines that the situation warrants such a degree of readiness.

Station telephones are available in the TSC for internal communication and communication to the OC and CMC. An outside telephone (independent of station switchboard) is available in the Unit 1 and 2 TSC. This telephone can be used with a computer terminal located in the same area. A plant telephone can be used for communication with the control room. However, due to the proximity of the TSC and Control Room, direct communication may be utilized. OPX network NRC phones are located in each TSC, in each Control Room and in the CMC. Radio base stations are established in the Unit 1 and 2 Control Room and in the OC or the CMC. Portable radios are available for communications with almost any other location in the station as well as with radiological monitoring teams in the environment.

Permanent area radiation monitors are installed in the Control Rooms. These provide local readout of radiation level and alarms if preset radiation levels are reached. Proximity of the Control Room and TSC allow these monitors to be used as an indication of the radiation levels in the TSC's. Each Control Room ventilation system employs gaseous radiation monitors to indicate and alarm on gaseous activity present in the Control Room. The TSC is served by the Control Room ventilation system so the Control Room gaseous radiation monitor also functions as the TSC monitor. Portable survey meters are available in the TSC's and in the OC.

Essential operating data from the unit's operator aid computers is available in the TSC's through video display, typewriter, line printer and through computer driven recorders. Approximately 2300 digital and 1600 analog points and numerous calculated values are available from each unit. All station drawings are available from a file room near the OC. Each TSC has drawing files which include system flow diagrams, one line electrical and instrument functional diagrams.

Since the Control Rooms share habitability systems with the TSC's procedures for evacuation of the TSC's to the Control Room are not applicable. If the OC becomes temporarily uninhabitable for any reason, the function to be carried out in that center can be performed in the TSC and Control Room areas.

A plan for upgrading the TSC to meet the total staffing and technical data requirements will be submitted by January 1, 1980.

c. Onsite Operational Support Center

An operational support staging area will be established to meet the concerns of this item by January 1, 1980.

ADDITIONAL ITEMS DESCRIBED IN NRC LETTER
OF
SEPTEMBER 13, 1979

1. Additional instrumentation requirements related to containment pressure, containment water level, and containment hydrogen monitors.

Containment Pressure Indication

In addition to the present existing qualified containment pressure transmitters, additional safety grade qualified pressure transmitters will be installed which are capable of displaying in the control room a range that includes a minus 5 psig to 3 times the concrete design pressure and 4 times the design pressure for steel.

Containment Water Level

In addition to the presently installed non-safety grade level instrumentation, the appropriate safety grade instrumentation to indicate in the control room the containment water level will be installed. This new instrumentation shall cover sufficient range to indicate an elevation equivalent to a 600,000 gallon capacity in the containment.

Containment Hydrogen Monitors

A design effort is underway to provide the capability to monitor the containment atmosphere for hydrogen. A system with safety grade hydrogen monitors capable of monitoring 0-10% hydrogen concentration will be provided with readout instrumentation in the control room. This instrumentation shall be operable with either a positive or negative pressure in the containment.

2. Remotely operable high point vent for gas from the reactor coolant system.

Duke Power Company has initiated a major design effort to provide the capability to remotely vent non-condensible gases in the Reactor Coolant System into the containment. The location and size of these vents will be designed to meet the objectives described in the staff letter. A description of the design will be provided by January 1, 1980.

3. Improvements in Emergency Procedures.

Duke Power Company has initiated an extensive review of our emergency preparedness program. The areas of concern within this item will be addressed by this review.