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ACCESSION NBR:8002060537 DOC.DATE: 80/01/31 NOTARIZED: NO DOCKET #
FACIL:50-269 Oconee Nuclear Station, Unit 1, Duke Power Co. 05000269
50-270 Oconee Nuclear Station, Unit 2, Duke Power Co. 05000270
50-287 Oconee Nuclear Station, Unit 3, Duke Power Co. 05000287

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DENTON,H.R. Office of Nuclear Reactor Regulation

SUBJECT: Forwards supplemental response to NUREG-0578 covering emergency power supply requirements & instrumentation for detection of inadequate core cooling.

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WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

January 31, 1980

TELEPHONE: AREA 704
373-4083

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:

Pursuant to an NRC Staff request during the Oconee site visit January 14-15, 1980, the attached information is provided to supplement my letters of October 18, November 21, and December 17, 1979. Supplemental information is provided for NUREG-0578 items 2.1.1, 2.1.3b, 2.1.4, 2.1.5c, 2.1.6a, 2.1.7b, 2.1.8a, and 2.1.8c. No additional information was requested for the other NUREG-0578 Category A items.

Very truly yours,

William O. Parker, Jr.

RLG:scs

Attachment

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DUKE POWER COMPANY
NUREG-0578 SUPPLEMENTAL RESPONSE
OCONEE NUCLEAR STATION

2.1.1 Emergency Power Supply Requirements

Pursuant to an NRC Staff request during the Oconee site visit January 14-15, 1980, the following information regarding minimum pressurizer heater requirements is provided:

The minimum pressurizer heater capacity required is 126 KW, which corresponds to a typical single bank of pressurizer heaters. This value was calculated by taking into account various heat losses.

The loss through the pressurizer insulation was calculated utilizing the outside surface area of the insulation and an average heat flux of 80 BTU/hr ft^2. This resulted in an approximate heat loss of 96,000 BTU/hr or 28 KW.

The loss through the uninsulated pressurizer areas around the horizontal heater bundles was calculated and resulted in an approximate heat loss of 50,000 BTU/hr or 15 KW.

Experience by Babcock and Wilcox, the Oconee NSSS vendor, has shown that the insulation losses may account for only 20-40% of the total losses. Therefore, a factor of 2.5 has been applied to the sum of the accounted losses. This results in an additional heat loss of 219,000 BTU/hr or 64 KW.

Thus, the total heat loss from the system is conservatively placed at 365,000 BTU/hr or 107 KW. One bank of heaters, 126 KW, is more than adequate to makeup for this loss.

Actual measured heat loss, which has been determined during initial startup testing, is on the order of 20-25 KW. Therefore, it is considered that the above calculations are valid.

Based on our review, the Oconee design fully meets the intent of this item.

2.1.3.b Instrumentation for Detection of Inadequate Core Cooling

Duke Power, along with other utilities with B&W nuclear supply system, requested that B&W investigate various means of detecting inadequate core cooling and provide recommendations for additional instrumentation for the detection of inadequate core cooling.

The investigation showed that the existing instrumentation could detect inadequate core cooling. The hot and cold leg RTD's along with pressure indication will indicate whether the RCS is subcooled

2.1.3.b (Continued)

or saturated. This determination will lead to operator actions designed to return the RCS to a subcooled condition. Should these actions be ineffective, the incore thermocouples or the hot leg RTD's indicating superheated temperatures for the existing RCS pressure will provide a positive indication that the core is partially uncovered and inadequately cooled. Those indications would lead to subsequent operator actions designed to depressurize the RCS to reach the low pressure injection system setpoint in order to reflood the core region. These operator actions will recover the core and begin to recover RCS inventory. The existing instrumentation will show when the core is recovered by a decrease in incore thermocouple readings.

Duke considers that existing instrumentation is adequate to detect inadequate core cooling.

B&W has recommended a hot leg level measurement system for reactor vessel level indication. Duke is reviewing this concept. In addition we are reviewing a method which utilizes radiation detectors. Duke intends to review both approaches to reactor vessel level measurement prior to accepting either. We are not convinced that either method is fully adequate.

Upon completion of our review, Duke will provide the information required by this item.

Attached are updated tables of information required on the subcooling meter. Data on computer availability during power operation will be provided by February 8, 1980.

2.1.4 Containment Isolation Provisions

During normal operation, Engineered Safeguards (ES) components are controlled from various points on the control panels, and the ES control module for each components in "AUTO." Upon actuation of an ES channel, each component is placed in the ES condition (e.g. pump-on, valve-shut, valve-open). An operator can take control of any component that has been actuated by ES signal, by changing the control module from "AUTO" to "MANUAL," and by controlling the component from the normal controls on the operating panel. If the control module is returned to "AUTO," the component is controlled by the ES actuation signal.

The following systems are isolated upon Engineered Safeguard actuation of either high Reactor Building pressure (4 psig) or low reactor coolant pressure (1500 psig):

1. Letdown line
2. Reactor Coolant Pump seal return line
3. Quench tank sample line
4. Quench tank gaseous vent
5. Reactor Building purge lines
6. Reactor Building sump drain line
7. Reactor Building atmosphere sample line

ATTACHMENT

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)	<1/2^{o}F, <1 PSI
Range of Display	-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 (Pol...
Continuous or on Demand	On-Demand
Single or Redundant Display	Single
Location of Display	Alarm CRT/Control Ro
Alarms (include setpoints)	See Description
Overall uncertainty (^{o}F, PSI)	<1/2^{o}F, <1 PSI
Range of Display	N/A
Qualifications (seismic, environmental, IEEE323)	N/A

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)	_ Hot leg/loop
Range of temperature sensors	20-650°F
Uncertainty of temperature sensors (°F at 1)	≤ 2°F
Qualifications (seismic, environmental, IEEE323)	Seismic & environment
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)	1-1/2% of span
Qualifications (seismic, environmental, IEEE323)	Seismic & environment.
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)

2.1.4 (Continued)

8. Pressurizer sample line
9. OTSG sample line
10. OTSG drain line

The following systems are isolated upon Engineered Safeguard actuation of high Reactor Building pressure (4 psig):

1. Component cooling to reactor coolant pumps
2. Low pressure service water to the reactor coolant pump motor

2.1.5c Shielding Requirements for Hydrogen Purge Installation

Additional information on this item will be provided by February 8, 1980.

2.1.6.a Integrity of Systems Outside Containment Likely to Contain Radioactive Materials

Additional information on this item will be provided by February 8, 1980.

2.1.7.b Auxiliary Feedwater Flow Indication

The emergency feedwater system at Oconee presently has one control grade transmitter installed in the flow path to each steam generator. These will be upgraded to safety grade. In addition to satisfy single failure criteria, a safety grade differential pressure transmitter sensing steam generator level will be added for each steam generator. The above modifications will be complete by January 1, 1981.

2.1.8a Improved Post-Accident Sampling Capability

Additional information on this item will be provided by February 8, 1980.

2.1.8c Improved Inplant Iodine Instrumentation

Additional information on this item will be provided by February 8, 1980.