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January 29, 1980

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

Attention: Mr. R. L. Baer, Chief
Light Water Reactor Project Branch No. 2

Re: McGuire Nuclear Station
Docket Nos. 50-369 and 50-370

Dear Mr. Denton:

My letter of November 19, 1979, transmitted a schedule for Duke Power Company's response to NUREG 0578 for McGuire Nuclear Station. Duke's initial response was transmitted by my letter of December 26, 1979. Attached is the second of Duke's three scheduled responses. Please note that any item discussed in this response supercedes our previous response.

Very truly yours,

William O. Parker, Jr.

William O. Parker, Jr. By [signature]

THH/sch
Attachment

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ADD:
M WILLIAMS
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DUKE POWER COMPANY

Response to NUREG 0578
Short Term Requirements for
McGuire Nuclear Station

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2.1.3a Indication of Relief and Safety Valves Position

PORV

The position of the pressurizer power-operated relief valves is detected by a seismically and environmentally qualified stem-mounted limit switch. The limit switch actuates indicator lights on the main control board. The entire circuit including power supply is safety-related. Additionally, a control room computer alarm is activated upon the opening of a PORV.

This PORV position indication is a feature of the current McGuire design.

Safety Valve

Flow through the safety valves is detected by an acoustic monitoring system. This system senses vibrations caused by the flow through the valve and translates this signal into an indication of valve position as a fraction of full open.

Flow induced vibrations are detected by accelerometers strapped to the safety valve piping. Two accelerometers will be located at each of the three valves. These signals are passed through a preamplifier to an electronic module located in the control complex area. The RMS value of this signal drives a bar graph which shows the valve position. The bar graph is a set of ten vertically arranged indicator lights which are labeled to give valve position as a fraction of full open.

An alarm is provided in the control room to indicate when the valve is not fully closed.

This system is safety-grade, meets the appropriate seismic and environmental qualification requirements, and will be installed prior to fuel loading.

2.1.3b <u>Instrumentation for Detection of Inadequate Core Cooling</u>

Degree-of-Subcooling Indication

The margin to saturation will be calculated from reactor coolant system pressure and temperature measurements (wide-range and low-range pressure and wide-range hot and cold leg temperature and temperature from in-core thermocouples). The thermocouple readings (approximately 60) are averaged and compared with the wide range RTD values. The highest of these temperatures and the lowest pressure are then used to calculate margin to saturation. Averaging of the thermocouple readings and calculation of margin to saturation are performed by the plant computer.

The computer output consists of a CRT graphic display of margin to saturation conditions, i.e., a plot of plant pressure and temperature in relation to a computer generated saturation curve. Additionally, this display also indicates in numerical terms RCS temperature, pressure, power level, margin to T_{sat}, and margin to P_{sat}. Alarm status is indicated by flashing the alarming parameter on the CRT display and by printout on the typewriter. Two alarm setpoints are provided for both T_{sat} and P_{sat}. The alarm setpoint is dependent on reactor power.

Normal control board instrumentation for RCS temperature and pressure will be used in conjunction with a control room copy of the steam tables and a written procedure to determine margin to saturation as a backup to the computer calculation.

This system for determining the degree of subcooling will be fully operational by fuel loading.

INFORMATION REQUIRED ON THE SUBCOOING METER

Display

Information Displayed (T-Tsat, Tsat, Press, etc.)	T-Tsat, P-Psat Temp., Press (Wide Range) % Power, Alarms
Display Type (Analog, Digital, CRT)	CRT
Continuous or on Demand	DEMAND
Single or Redundant Display	SINGLE
Location of Display	CONTROL ROOM
Alarms (include setpoints) Setpoints:	>0% FP <0% FP 30°F 50°F 430 PSIA 600PSIA
Overall uncertainty (°F, PSI)	.200 to .750 PSIA .175 to .575°F
Range of Display	PROGRAMMABLE
Qualifications (seismic, environmental, IEEE279)	N/A

Calculator

Type (process computer, dedicated digital or analog calc.)	HONEYWELL HS4400 PROCESS COMPUTER
If process computer is used, specify availability. (% of time)	99.21% (1979 Average)
Single or redundant calculators	SINGLE
Selection Logic (highest T., lowest press)	HIGHEST VALID TEMPERATURE LOWEST VALID PRESSURE
Qualifications (seismic, environmental, IEEE279)	N/A
Calculational Technique (Steam Tables, Functional Fit, ranges)	STEAM TABLES (1967 ASME)

Input

Temperature (RTD's or T/C's)	T/C & RTD
Temperature (number of sensors and locations)	Approx. 60 IN-CORE T/C; Two wide range RTD's per loop
Range of temperature sensors	T/C : 0-2300°F RTD : 0-700°F

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Uncertainty* of temperature sensors (°F at 1σ)	<2.0°F T/C ≤1.5°F RTD
Qualifications (seismic, environmental, IEEE279)	RTDs (seismic, environmental) T/Cs (none)
Pressure (specify instrument used)	RCS wide range press. RCS low range press.
Pressure (number of sensors and locations)	2-Reactor Coolant System
Range of Pressure sensors	Low Range 0-800 PSIG Wide Range 0-3000 PSIG
Uncertainty* of pressure sensors (PSI at 1σ)	±1% span
Qualifications (seismic, environmental, IEEE279)	Wide range (seismic, environmental Low range (none)

Backup Capability	INCORE T/C-CONTROL ROOM METER WITH SELECTOR SW. HOT AND COLD LEG TEMP. (RTDs)-CONTROL ROOM RECORDER PRESSURE-CONTROL ROOM METER AND 1 CHANNEL RECORDED
Availability of Temp & Press	
Availability of Steam Tables etc.	Copy available in control room.
Training of operators	Yes
Procedures	Yes

*Uncertainties must address conditions of forced flow and natural circulation

2.1.6a Systems Integrity for High Radioactivity

A periodic leak rate test will be written for systems carrying radioactive fluids outside of containment. This test, to be performed during each refueling outage, will be accomplished by pressurizing a system or part of a system and checking non-welded pipe joints, penetrations, flanges, valve separations, packing, and pump packing for leakage. Where possible, pumps included in the leak test boundary will be run so that a more accurate determination of the leak rate may be made.

A separate periodic test procedure will be written to assure that any leakage is detected on a timely basis. This test will be run at least daily and will require that systems carrying radioactive fluids outside of containment be visually inspected for excessive leakage. Appropriate corrective action will be taken if any leakage is detected.

2.1.6b Plant Shielding Review

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 McGuire Nuclear Station has been initiated to determine if any areas of the station fail 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 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 McGuire Nuclear 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, water injected by the Safety Injection System, and water from the Ice Condenser melt.)

To aid in identifying potential personnel access problems, the station will be divided into post-LOCA radiation zones. Systems to be considered in

2.1.6b (con't)

determining the post-LOCA radiation zones will be: residual heat removal, recirculation, letdown, and radwaste. The major emphasis of the review will be to assure that station personnel would be able to carry out their emergency procedures. Upon completion of this review, scheduled for April 1, 1980, Duke will submit a schedule for implementing any required design changes.

In addition to determining radiation zones, integrated exposures will be calculated for use in evaluating equipment radiation qualification. A location specific review will be conducted 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. The equipment qualification review will be completed following the resolution of any personnel access concerns.

2.1.8a Post Accident Sampling Capability

As stated in the response to Item 2.1.6b, a station review is in progress to determine post-accident radiation levels and shielding adequacy. This review, scheduled to be completed by April 1, 1980, will identify any necessary design changes to the sampling areas. Following evaluation of this review, Duke Power Company will submit a description of any design changes to be implemented.

Procedures will provide for prompt radiological spectrum analyses of noble gases, radioiodines, radiocesiums, and other nonvolatile radionuclides. Included is a boron analysis procedure capable of being performed within one hour. 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.

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

Vent monitors for noble gases will be provided with a range adequate to cover normal and anticipated conditions. Three monitors will be required to measure activities from $1x10^{-7}$ µCi/cc to $1x10^{5}$ µCi/cc of noble gases with one decade overlap between each monitor. The monitors are qualified to IEEE-323, 1971. Continuous indication and recording of the monitors will be provided in the control room. Delivery of these monitors is expected by March 1, 1981 with installation by May 1, 1981.

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.

Two physically and electrically separated containment radiation monitors will be provided to monitor radiation levels up to 10^{8} Rad/hr. These monitors will be qualified to IEEE-323, 1971 and powered from the vital instrument bus. Monitor output will be indicated and recorded continuously in the control room. The monitors will be installed prior to full power operation and shall be calibrated as indicated above for the vent monitors at refueling as required by the Technical Specifications.

Procedures will be developed to quantify releases from the unit vents, waste gas decay tanks, main condenser air ejector, and the Auxiliary Building.

2.1.8c Improved In-Plant Iodine Instrumentation

Silver Zeolite radioiodine sampling cartridges are in use at McGuire for sampling air when the presence of noble gases is suspected. McGuire Health Physics personnel are knowledgeable in the appropriate station procedures required and are trained in the equipment required to determine airborne iodine concentrations in the plant under all conditions. Procedures to determine airborne iodine concentrations will cover analyses to be done if counting room capabilities are not available.

2.1.9 <u>Transients and Accident Analysis</u>

1) <u>Small Break LOCAs</u>

2) <u>Inadequate Core Cooling</u>

Duke Power Company is in the process of developing new procedures and training guidelines for dealing with small break loss-of-coolant accidents and incidents of inadequate core cooling. This effort is based on analyses conducted by Westinghouse Electric Corporation.

Westinghouse has completed its analysis of small break loss-of-coolant accidents for Upper Head Injection plants. This analysis has been submitted to the NRC in WCAP 9600 and WCAP 9639. Duke is currently reviewing this analysis and will make any necessary changes in procedures and training guidelines before fuel loading.

Westinghouse has also submitted an analysis of inadequate core cooling to the NRC. An additional analysis is currently scheduled for submittal to the NRC by March 31, 1980. Duke will assure that procedures and training guidelines are consistent with both of these analyses before fuel loading.

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2.2.1b Shift Technical Advisor

The two functions of the Shift Technical Advisor, 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 fuel loading. It is intended that he will provide the on-shift accident assessment capability. Further training will be conducted 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 prior to fuel loading.

2.2.2a Control Room Access and Authority Succession

Administrative procedures have been written to limit personnel access to the control room and to establish a clear line of authority for coping with operational transients and accidents. The McGuire Security Plan controls access to all vital areas of the plant including the control area. In addition, Station Directive 3.1.4, Conduct of Operations, has been written to control access to and actions within the area designated as "Surveillance Area" in FSAR Figure 13.5.1-1.

2.2.2b <u>On-Site Technical Support Center</u>

Duke Power Company is in the process of establishing an onsite Technical Support Center at McGuire Nuclear Station to serve both units in an area on the same elevation and 40' south of the control room. Plans for this area include:

a. Ready access to as-built plant drawings including general arrangement drawings, flow diagrams, electrical and instrument drawings by fuel loading.

b. Installation of a computer terminal having the capability to access, print and/or display plant parameters independent from control room actions by January 1, 1981. The functional equivalent will be provided by fuel loading by using existing computer terminals.

c. Provisions for habitability to the same degree as the control room for postulated accident conditions. This will include the installation of an iodine filter system by January 1, 1981.

d. Establishment of dedicated communications capability with the control room, the offsite Crisis Management Center and with the NRC by fuel loading.

e. Installation of monitoring equipment which will provide local readout of radiation level and alarms if preset radiation levels are reached. This installation will be complete by January 1, 1981. Portable radiation survey instruments will be available in the Technical Support Center by fuel loading.

The Technical Support Center includes approximately 1200 square feet of space. It is in close proximity to the control room and has similar environmental control features. Procedures will be developed for performing the accident assessment function from the control room.

The anticipated completion date for all functional items in this center is January 1, 1981, contingent upon timely equipment delivery. Procedures being prepared now include the identification of this area as the Technical Support Center. As new portions of the facility become available, they will be included in future procedure revisions. The Technical Support Center will be established prior to fuel loading and will be upgraded to meet all applicable requirements by January 1, 1981.

Staffing of the Technical Support Center includes the Station Manager, Superintendent of Operations, Superintendent of Technical Services, Superintendent of Maintenance, Superintendent of Administration, Station Health Physicist and staff personnel necessary to support them.

2.2.2c <u>On-Site Operational Support Center</u>

An area adjacent to the control room has been designated as the Operational Support Center. Direct communication with the control room is provided, and procedures will be written to govern its use in emergency [illegible]ations. The McGuire emergency plan will be revised to reflect the existence of the center and to establish the methods for utilization of this area. The Operational Support Center will be fully established prior to fuel loading.

A-1 Containment Pressure

Continuous indication of containment pressure will be provided in the control room. Measurement and indication range will extend from minus five psig to four times the design pressure of the Containment. The design will meet the guidelines of NRC Regulatory Guide 1.97, including qualification, redundancy, and testability.

Unit 1 installation is scheduled to be complete by January 1, 1981. Unit 2 installation will be completed by fuel loading. These schedules are contingent upon delivery of required instrumentation components.

A-2 Containment Water Level

Two containment floor and equipment sumps are provided on the floor of the lower containment (El 725') to collect floor drains and equipment drains. However, these sumps and their associated pumps and instrumentation serve no safety function.

The containment emergency recirculation sump at McGuire encompasses the entire floor of the lower containment. The two ECCS recirculation lines take suction just inside the Containment wall at elevation 725' and are oriented horizontally. They are not located in the bottom of a recess or sump in the floor. Redundant safety grade level instrumentation is provided to measure emergency recirculation sump level. The range of this instrumentation is 0-20 feet (El 725' to El 745') which is equivalent to a lower containment volume of approximately 1,000,000 gallons. The accuracy of this instrumentation is ±10% over the full range.

The McGuire containment emergency sump level instrumentation provides the required narrow range and wide range level measurement functions. This instrumentation is currently being evaluated to determine if it meets the requirements of Mr. H. R. Denton's letter of October 30, 1979. Where necessary, Duke will modify this instrumentation to conform to the above requirements. All modifications should be complete by January 1, 1981.

A-3 Hydrogen Monitoring

Continuous indication of hydrogen concentration in the containment atmosphere will be provided in the control room. Measurement capability will be provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure. The design of this instrumentation will meet the guidelines of Regulatory Guide 1.97 including qualification, redundancy, and testability.

This instrumentation is scheduled to be installed on Unit 1 by January 1, 1981 and on Unit 2 by fuel loading. These schedules are contingent upon delivery of the required instrumentation components.

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A-4 RCS Venting

Duke Power Company will provide remotely operable Reactor Coolant System and reactor vessel head high point vents which are safety grade, satisfy the single failure criterion, meet the requirements of IEEE-279, and other requirements put forth on H. R. Denton's letter of October 30, 1979.

Installation is scheduled to be complete by January 1, 1981 for Unit 1 and by fuel loading for Unit 2. These schedules are contingent upon delivery of required equipment.