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September 8, 1980

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

Attention: Mr. B. J. Youngblood
Licensing Projects Branch No. 1

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

Dear Mr. Denton:

Enclosed with this letter are forty copies of updated responses to the document "Duke Power Company, McGuire Nuclear Station, Response to TMI Concerns." This document was transmitted to the NRC via my letter of May 23, 1980 and updated via my letters of July 18, 1980 and August 6, 1980.

Duke's response to the information requested by August 1, 1980 in Mr. H. R. Denton's letter of March 28, 1980 to All Power Reactor Applicants and Licensees is provided in this document. This response is contained in the document sections addressing the McGuire operator training programs.

Fuel loading for McGuire Unit 1 is currently scheduled for October 1980. Please schedule your review of this document accordingly.

Very truly yours,

William O. Parker, Jr.

THH:scs
Enclosures (40)

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WILLIAM O. PARKER, JR., being duly sworn, states that he is Vice President of Duke Power Company; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this document, Duke Power Company, McGuire Nuclear Station Response to TMI Concerns, and that all statements and matters set forth therein are true and correct to the best of his knowledge.

William O. Parker, Jr., Vice President

Subscribed and sworn to before me this 8th day of September, 1980.

Notary Public

My Commission Expires:

September 20, 1984

DUKE POWER COMPANY
MCGUIRE NUCLEAR STATION

Response to TMI Concerns
September 8, 1980

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Appendix A McGuire Nuclear Station Procedures

Station Directive 3.1.4, Conduct of Operations
Station Directive 3.8.2, Station Emergency Organization
Station Directive 3.1.9, Relief at Duties of Plant Operation
Periodic Test PT/1/A/4700/10, Shift Turnover Verification
Station Directive 3.1.31, Duties, Responsibilities and Qualifications of the Shift Technical Advisor
Station Directive 3.1.32, Station Safety Engineering Group

Appendix B Control Room Design

August 15, 1980 letter from Mr. W. O. Parker to Mr. H. R. Denton

Appendix C NRC Requirements for McGuire Nuclear Station

September 27, 1979 letter from D. B. Vassallo to all Pending Operating License Applicants
November 9, 1979 letter from D. B. Vassallo to all Pending Operating License Applicants
March 28, 1980 letter from H. R. Denton to all Power Reactor Applicants and Licensees
NUREG-0694: TMI-Related Requirements for New Operating Licenses

Appendix D NRC Requests for Additional Information

June 4, 1980 letter from B. J. Youngblood to W. O. Parker
June 30, 1980 letter from B. J. Youngblood to W. O. Parker
July 2, 1980 letter from B. J. Youngblood to W. O. Parker
July 23, 1980 letter from R. L. Tedesco to W. O. Parker
August 25, 1980 letter from R. L. Tedesco to W. O. Parker

The event investigator prepares a report describing the cause of the event and any relevant plant behavior, and outlining proposed corrective actions. The SSEG then reviews each report for accuracy and completeness, and assesses the adequacy of proposed corrective actions. The SSEG submits its report to the Station Manager and the NSRB for review and for approval of corrective actions, to the Shift Technical Advisor for review with regard to operating procedures, and to the Supervisor of Training for inclusion of relevant information in the training program.

The SSEG will be composed of four full time engineers assigned to the day shift. The group will be staffed on a rotating basis from among experienced station personnel and will be multidisciplined with expertise in the areas of instrumentation, maintenance, operations, and technical services. Additional information on the membership and duties of the SSEG is provided in Station Directive 3.1.32 which is included in Appendix A.

Upon notification of an event, the General Office PC&L Section notifies company management, and may alert other General Office engineering and scientific support groups. The Station Manager forwards the approved event report to the General Office PC&L Section, where a Licensee Event Report (LER) is prepared and submitted, if necessary. Information is provided to other organizations, including NSAC, NRC, and the NSSS vendor. Detailed evaluations of plant transients are performed, and event occurrence data is maintained. As appropriate, other engineering support groups review the LER and station event reports for further recommendations on corrective actions, and may interface with appropriate equipment vendors. The NSRB performs an independent review of the event report, the LER, and the effectiveness of any follow-up actions.

INDUSTRY EXPERIENCE EVALUATION

Figure 2 illustrates the flow path for information received concerning industry operating experience. Significant events will be brought to the attention of Duke Power Company by NSAC, NSSS vendors, other utilities, or the NRC. Information is distributed, as appropriate, to General Office engineering support groups for review and development of corrective actions and to the Training Services group for incorporation into the training program. The SSEG reviews the information for applicability ot the specific station, and makes recommendations to the NSRB and the Station Manager in areas where action may be necessary. The Station Manager then developes and implements appropriate corrective actions with assistance from and review by the engineering support groups.

REVISED SCOPE AND CRITERIA FOR LICENSING EXAMINATIONS

Reference: Action Plan - Plan I.A.3.1

The McGuire reactor operator and senior reactor operator license applicants have taken the NRC pre-revision written examinations. However, these applicants were required to pass the examinations with a grade of 80% overall and 70% in each category. Those license applicants who must be reexamined will take the new revised examination. The McGuire license applicants have submitted letters to Mr. Paul Collins, NRC Operator Licensing Branch, which request the release of their examination results.

The McGuire license requalification program will include academic instruction in heat transfer, fluid flow, thermodynamics, and mitigation of accidents involving a degraded core. The program will include both normal and emergency operation instruction on the McGuire simulator and will meet the requirements of Enclosure 4 to Mr. H. R. Denton's letter of March 28, 1980 to All Power Reactor Applicants and Licensees.

ADMINISTRATION OF TRAINING PROGRAMS FOR LICENSED OPERATORS

Reference: Action Plan - I.A.2.3

Training instructors who teach systems, integrated responses, transient, and simulator courses at McGuire either have previously held an SRO license or will successfully complete a SRO examination. Applications for those instructors who were prepared for this examination were submitted by August 1, 1980. The remaining instructors will function under the supervision of an instructor who has previously held a SRO license until they are sufficiently prepared to apply for the SRO examination. It is the policy of Duke Power Company that all training instructors are informed of plant modifications, procedural changes, and current station administrative policies.

TRAINING FOR MITIGATING CORE DAMAGE

Reference: Action Plan - II.B.4

Duke has modified the McGuire training program in order to place increased emphasis on the operation and significance of any McGuire systems or instrumentation which could be used to monitor and control accidents in which the core may be severely damaged. This additional training identifies the vital instrumentation which supplies the operator with needed information in a degraded core situation. The training also identifies alternate methods of obtaining this information as well as specific instruction in the interpretation of instrument readings in degraded core situations.

The McGuire operators currently enrolled in the operator training program will receive the training for mitigating core damage at the end of the program. These operators will be required to apply their knowledge of plant operation to hypothetical degraded core situations. Various degraded core scenarios will be presented which will require the operators to diagnose plant status and restore the plant to a safe condition utilizing both primary and alternate instrumentation as their source of information.

Currently licensed McGuire operators will receive this training in a special supplemental training class. Each operating shift will have received this training prior to power escalation of Unit 1. The training for these operators will be similar to that described above. It will, however, include a review of their prior training with increased emphasis placed upon degraded core situations.

Training for mitigating core damage will be incorporated into the operator training program for all future McGuire license applicants. In addition, selected Duke training instructors will participate in a degraded core training seminar sponsored by General Physics Corporation in early October.

The existing body of knowledge regarding nuclear plant response under degraded core conditions is being enlarged. Duke is participating in this effort in conjunction with other utilities, INPO, and the NSSS vendors. Information resulting from this effort will be incorporated into the McGuire operator training and requalification programs soon after it is available. Enclosure 3 to Mr. H. R. Denton's letter of March 28, 1980 to All Power Reactor Applicants and Licensees is being used as a basis in the development of this information. In addition the cleanup effort at TMI Unit 2 should provide significant information in this regard.

TRAINING DURING LOW POWER TESTING

Reference: Action Plan - I.G.1

Duke Power Company will conduct a series of special low power tests at McGuire Nuclear Station for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental operator training. These tests will be performed at reactor power levels no greater than five percent of full power.

The low power tests to be performed at McGuire are listed below:

1. Natural Circulation Verification
2. Effect of Steam Generator Isolation on Natural Circulation
3. Natural Circulation with Simulated Loss of Offsite Power
4. Natural Circulation at Reduced Pressure
5. Natural Circulation with Loss of Pressurizer Heaters
6. Simulated Loss of All Onsite and Offsite AC Power

Each operating shift will either observe or participate in tests 1, 4, and 5. Tests 2, 3, and 6 are designed to provide plant specific technical information and therefore will only be performed once.

One page abstracts of each of these tests are provided in the pages which follow. The complete test procedures are being developed and will be reviewed by Westinghouse. A safety analysis of this program is being prepared by Westinghouse and will be submitted for NRC review by mid October 1980.

Duke does not intend to perform the "Cooldown Capability of the Charging and Letdown System" (CC/ 3) test at McGuire. The purpose of this test is to demonstrate the capability of the Chemical and Volume Control System (CVCS) to cooldown the reactor coolant system (RCS) with the steam generators isolated while using RCS pump heat as the heat source. This test has very little value from either a technical information collection standpoint or an operator training standpoint.

The energy removal capability of the CVCS has been long recognized by the operators at McGuire. During normal operation, the CVCS removes approximately 3 MW from the primary system. The operators have had many opportunities to experience the effects of varying letdown rates on the primary system during the approximately 10 weeks of Hot Functional Testing performed at McGuire. Therefore, a significant training benefit will not result from performing this test.

The ability to cool the primary system has been verified at McGuire during the CVCS functional test. The heat removal capacity of the McGuire CVCS is less than 5 MW which is extremely small in comparison to the heat loads normally experienced shortly after shutdown. This capacity is approximately equal to the decay heat load normally experienced by McGuire's core after approximately one hundred days. In the unlikely event that the McGuire CVCS would be required for cooldown after this length of time, there would be adequate time to prepare for such a cooldown.

In addition, the CCCLS test has been recently performed at Sequoyah Unit 1, North Anna Unit 2, and Salem Unit 2. The results of these tests, which are applicable to McGuire, have confirmed the existing knowledge regarding primary system cooldown using the CVCS.

The "Establishment of Natural Circulation from Stagnant Conditions" test was recently performed at Sequoyah Unit 1. Selected McGuire personnel observed the performance of this test at Sequoyah (the entire Sequoyah low power test program was observed). The results of this test are being evaluated by Duke and the appropriate portions will be incorporated into the McGuire simulator for future training.

The "Boron Mixing and Cooldown" test was performed at Sequoyah with the reactor critical. This test will be performed at Diablo Canyon Unit 1 using decay heat after completion of the power ascension program and manufacturer's acceptance test. The thermohydraulic properties of Sequoyah, Diablo Canyon, and McGuire are extremely similar. Therefore, the test results from Sequoyah and Diablo Canyon will be applicable to McGuire. Duke is currently evaluating the Sequoyah test results and will evaluate the Diablo Canyon test results, and the appropriate results will be incorporated into the McGuire operator training program.

The above deviations from the low power test program performed at Sequoyah do not compromise Duke's low power test program for McGuire. This McGuire program satisfies the NRC requirement for low power tests to provide meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental operator training. Appropriate information resulting from this McGuire program will be incorporated into the McGuire operator training program.

Natural Circulation Verification

Purpose

Verify that natural circulation is established in the primary system following the loss of forced reactor coolant flow.

Prerequisites

Reactor Power $\leq 3\%$
Normal primary system temperature and pressure.
Reactor Coolant Pumps operating.
Pressurizer Heaters controlling pressure.
Steam Generators being fed by normal feedwater.

Test Method

The test will be initiated by tripping all reactor coolant pumps. The establishment of natural circulation will be verified by observing the response of the hot and cold leg temperature instrumentation in each loop.

Acceptance Criteria

Natural circulation is successfully established in all four reactor coolant loops.

Effect of Steam Generator Isolation on Natural Circulation

Purpose

To observe the effects of steam generator isolation on natural circulation and to verify that these effects do not adversely affect the performance of the operable primary loops.

Prerequisites

Reactor Power $\leq$1%
Primary system temperature ~515°F and pressure ~2000 psig.
Reactor Coolant Pumps shutdown.
Pressurizer Heaters controlling pressure.
Steam Generators being fed by normal feedwater.
Natural Circulation established in all 4 loops.

Test Method

With natural circulation successfully established in all four primary loops, cool the reactor coolant system down to provide sufficient margin to the steam generator safeties. Isolate steam generators one at a time while adjusting feed flow to the unisolated steam generators to maintain a coolant primary system average temperature. Continue this until two steam generators remain unisolated or until the average primary system temperature can no longer be prevented from increasing. The hot and cold leg temperatures will be observed to ensure the core is being sufficiently cooled by the natural circulation process. The steam generators will then be returned to service one at a time and natural circulation will be verified to reestablish in each loop.

Acceptance Criteria

1. Natural circulation in the unisolated loops is not interrupted or significantly impaired by the isolation of steam generators on other loops.

2. Natural circulation can successfully be reestablished on the previously isolated loops.

Natural Circulation with Simulated Loss of Offsite Power

Purpose

Verify that natural circulation cooling can be maintained following a loss of offsite power.

Prerequisites

Reactor Power $\leq$1%
Normal primary system temperature and pressure.
Reactor Coolant Pumps operating.
Pressurizer Heaters controlling pressure.
Steam Generators being fed by auxiliary feedwater.

Test Method

The test will be initiated by simulating a loss of offsite power. The reactor coolant pumps will be tripped and the auxiliary feed pump and pressurizer heater loads will be transferred to diesel power. Natural circulation will be verified by observing the response of the hot and cold leg temperature instrumentation on each loop.

Acceptance Criteria

Natural circulation is successfully reinstated in all four loops following the interruption of feedwater flow to the steam generators.

Natural Circulation at Reduced Pressure

Purpose

1. To provide Operations personnel with online experience in using the Operator Aid Computer Saturation Monitor Program to monitor and control margin to saturation.

2. To verify that changes in saturation margin will not affect natural circulation provided that adequate margin to saturation exists.

Prerequisites

Reactor Power $\leq$3%
Normal primary system temperature and pressure.
Reactor Coolant Pumps operating.
Pressurizer Heaters controlling pressure.
Steam Generators being fed by normal feedwater.

Test Method

The test will be initiated by tripping all reactor coolant pumps and verifying the establishement of natural circulation. As the primary system temperature is reduced, the primary system pressure will be reduced. The Saturation Monitor Program results will be monitored and compared against hand calculations. The effect of pressure reduction on natural circulation will be observed.

Acceptance Criteria

1. Natural Circulation is successfully established in all four reactor coolant loops.

2. The Operator Aid Computer Saturation Monitor Program provides reliable information concerning the margin to saturation.

3. Reducing the margin to saturation does not impair natural circulation as long as sufficient margin exists.

Natural Circulation with Loss of Pressurizer Heaters

Purpose

Verify establishment of natural circulation and determine the rate of decrease of margin to saturation while in this mode and the ability to reestablish margin through cooldown and makeup.

Prerequisites

Reactor Power $\leq$3%
Normal primary system temperature and pressure.
Reactor Coolant Pumps operating.
Pressurizer Heaters controlling pressure.
Steam Generators being fed by normal feedwater.

Test Method

The test will be initiated by tripping all reactor coolant pumps and the pressurizer heaters. Establishment of natural circulation will be verified by observing the response of the hot and cold leg temperature instrumentation in each loop. Operations personnel will observe the Saturation Monitor to assure margin. Prior to reaching saturation, secondary side steam flow will be increased to affect a cooldown and a reestablishment of saturation margin will be verified. Any shrinkage associated with cooldown will be compensated for by reactor water makeup.

Acceptance Criteria

1. Natural circulation is successfully established in all four reactor coolant loops.

2. Natural circulation is successfully maintained during the depressurization associated with the loss of pressurizer heaters.

3. Margin to saturation is successfully reestablished following the depressurization associated with the loss of pressurizer heaters.

TP/1/A/2150/26

SIMULATED LOSS OF ALL
ONSITE AND OFFSITE AC POWER

Purpose

1. To demonstrate that auxiliary feedwater can be controlled by manual means to remove heat energy from the primary system in order to maintain hot standby conditions.

2. To demonstrate that critical plant operations can be performed using emergency lighting.

3. To demonstrate the ability of 125 volt battery to supply the 125 volt vital AC.

4. To demonstrate that selected equipment areas do not exceed maximum design temperatures.

Prerequisites

Reactor subcritical by $\geq$ 1.6% $\Delta k/k$.
Normal primary system temperature and pressure.
Reactor Coolant Pumps operating.
Pressurizer Heaters controlling pressure.
Steam Generators being fed by auxiliary feedwater.

Test Method

Test will be initiated by simultaneously performing the following:

1. Tripping the pressurizer heaters.
2. Removing AC power from the auxiliary feedwater components.
3. Tripping selected space and equipment coolers.
4. Tripping vital battery chargers and AC power to inverter.
5. Isolating main feedwater and main steam lines.

Operations personnel will then establish manual control of the auxiliary feedwater system and provide sufficient flow to the steam generators to maintain a constant primary system temperature. These conditions shall be maintained for one hour after which AC power will be restored and equipment returned to normal service.

Acceptance Criteria

1. Emergency lighting in the station is sufficient to operate critical equipment with the loss of all normal lighting.
2. Hot standby conditions can be maintained for a 1-hour period with critical equipment operating off of vital battery power.
3. Manual operation of auxiliary feedwater valves and main steam power reliefs can be coordinated by the operators.
4. Cirtical equipment areas do not exceed design temperature limits.

ACCIDENT ANALYSIS AND PROCEDURE REVISION

References: NUREG-0578 - 2.1.9
Action Plan - I.C.1

Duke is in the process of developing new procedures and training guidelines for controlling and mitigating small break LOCAs, incidents of inadequate core cooling, and certain anticipated transients. Duke's effort is in conjunction with analysis and research being performed by Westinghouse.

The Westinghouse analysis of small break LOCAs in upper head injection plants, WCAP 9600 and WCAP 9639, has been submitted to the NRC for their review. Duke has reviewed these reports and made the necessary modifications to the McGuire emergency procedures and training program.

The Westinghouse analysis of inadequate core cooling, WCAP 9753, WCAP 9754, and WCAP 9744, has been submitted to the NRC for their review. These reports provide an analytical basis for subsequent Westinghouse development of guidelines for the detection of and recovery from inadequate core cooling. Duke will assure that the McGuire emergency procedures and training program are consistent with these forthcoming Westinghouse guidelines. Westinghouse has proposed a preliminary guideline which provides an interface between the emergency procedures and the interim inadequate core cooling instructions. This guideline was submitted to the NRC by the Westinghouse Owners Group on July 15, 1980. Duke is currently reviewing this guideline and will incorporate the appropriate provisions into the McGuire interim inadequate core cooling instructions.

The Westinghouse analysis of selected transients and accidents is continuing on a deliberate schedule. Duke is closely following the development of this analysis and will modify the McGuire emergency procedures and training program as appropriate.

CONTROL ROOM DESIGN

Reference: Action Plan - I.D.1

A detailed design review has been instituted for the McGuire control room to:

(A) Evaluate the amount, type, and form of information available to the unit operator.

(B) Review the requirements for control by the operator, and assess the degree to which the design of the control boards enables the operator to perform (or constrains him from performing) his duties safely and efficiently, and minimizing the potential for operator error.

(C) Identify and implement those changes and modifications to equipment, its arrangement and identification, on a schedule addressing the concerns for safety and operability.

This review has revealed two classifications of corrective actions, those to be implemented before fuel loading, and those to be implemented before startup following the first refueling. A human factors engineering design review of the McGuire control room was conducted during the week of June 2-6, 1980 by the NRC/Human Factors Engineering Branch. Duke evaluated the review team's report and responded to the NRC via Mr. W. O. Parker's letter of August 15, 1980. This response is provided in Appendix B.

RELIEF AND SAFETY VALVE POSITION INDICATION

References: NUREG-0578 - 2.1.3a
Action Plan - II.D.3

PORV

The position of the pressurizer power-operated relief valves is detected by seismically and environmentally qualified stem-mounted limit switches. The limit switches actuate 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.

Safety Valve

Flow through the safety valves is detected by an acoustic flow detection system. This system senses vibrations caused by flow through the valve which is an indication that the valve is not fully closed.

Two accelerometers have been strapped to the discharge piping of each safety valve. One of these is an installed spare and is wired to the electronics cabinet but not monitored. A charge converter processes the accelerometer output and provides the voltage input to the monitor. The RMS of this signal is related to the flow through the valve. This signal is filtered and amplified and is available on a front panel BNC connector. An RMS to DC converter provides on output to drive a bar graph on the front panel. 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. The charge converter is located in containment and the electronics cabinet is in the electrical penetration room.

The alarm output of the monitor is used to provide indication and alarm when flow exists through any of the three safety valves. A safety grade indicator light and a non-safety annunciator are provided. The bar graphs on the monitor can be used to determine which valve is open.

The system, with the exception of the annunciator alarms, is safety-grade, meets the appropriate seismic and environmental qualification requirements, and has been installed.

AUXILIARY FEEDWATER INITIATION AND INDICATION

References: NUREG-0578 - 2.1.7a and 2.1.7b
Action Plan - II.E.1.2

Automatic Initiation

Safety-grade automatic initiation and safety-grade emergency power for the Auxiliary Feedwater System are features of the McGuire Nuclear Station design (reference FSAR Ch. 10).

The automatic initiation circuitry for the Auxiliary Feedwater System meets the single failure criteria. Additionally, for most failures which could prevent the automatic start of an individual auxiliary feedwater pump, manual initiation of the affected pump is available from the Control Room. However, should the auxiliary feedwater pump in one safety train not be available due to any single failure, the redundant safety train is available with no loss of system function.

In the final stages of plant shutdown, the motor-driven auxiliary feedwater pumps must be tripped. For this reason the automatic auxiliary feedwater pump start upon trip of both main feedwater pumps or steam generator low-low level must be bypassed. This bypass is accomplished manually by means of a bypass switch located in the Control Room. This bypass is administratively controlled by use of operating procedures. When the bypass switch is in the bypass position, an annunciator is actuated and a status light is illuminated on the bypass status panel as required by Regulatory Guide 1.47.

The turbine-driven auxiliary feedwater pump does not have a bypass feature.

Indication

Safety grade indication of auxiliary feedwater flow to each steam generator has been provided in the McGuire control room. Provisions for calibration and testing were incorporated into the design of this instrumentation.

Control grade flow instrumentation in the lines to each steam generator and in the suction piping to each auxiliary feedwater pump is also provided. This control grade flow instrumentation is powered from the highly reliable battery-backed 120 VAC Auxiliary Control Power System (FSAR Section 8.3.2.1.3). Provisions for calibration and testing are included in the design of this control grade flow instrumentation.

AUXILIARY FEEDWATER SYSTEM RELIABILITY EVALUATION

Reference: Action Plan - II.E.1.1

An evaluation of the McGuire auxiliary feedwater system has been performed by Duke and Westinghouse. This evaluation consists of the following items:

1. a simplified auxiliary feedwater system reliability analysis that uses event-tree and fault-tree logic techniques to determine the potential for AFWS failure following a main feedwater transient, with particular emphasis on potential failures resulting from human errors, common causes, single point vulnerability, and test and maintenance outage;

2. a determination of the extent to which the McGuire auxiliary feedwater system meets each requirement in Standard Review Plan 10.4.9 and Branch Technical Position ASB-10-1; and

3. a determination of the design basis for the McGuire auxiliary feedwater system flow requirements and verification that these requirements are met.

Mr. W. O. Parker's letter of August 13, 1980 to Mr. H. R. Denton transmitted this evaluation. It revealed that no modifications to the McGuire auxiliary feedwater system are necessary.

REACTOR COOLANT SYSTEM VENTS

Reference: Action Plan - II.B.1

Duke has installed a reactor vessel head high point vent that is remotely operable from the McGuire control room. A one-inch line has been added to the existing reactor vessel manual vent line with the connection located before the first isolation valve. The new vent line contains two parallel flow paths with redundant fail closed solenoid valves in each flow path. The valves have been designed to pass non-condensible gases, water, steam, and mixtures thereof. Under normal operation these valves are deenergized. Valve position is indicated in the control room. Train A emergency power serves both isolation valves in one flow path, and Train B emergency power serves both isolation valves in the parallel flow path. A flow limiting orifice has been installed in the common line downstream of the isolation valves.

The McGuire Reactor Coolant System (RCS) vent system is safety grade, seismically qualified, meets the requirements of IEEE 279-1971, and satisfies single failure criteria. This vent system will be operable before fuel loading.

As mentioned above, this head vent system has been designed to single failure criteria. If any single failure prevents a venting operation through one flow path the second flow path is available for venting the RCS. The two isolation valves in each flow path provide a single failure method of isolating RCS venting.

Inadvertent actuation of RCS venting is limited by the use of fail closed solenoid isolation valves. In addition the use of an orifice in the common line downstream of the valves limits the flow to less than the make up capacity of the RCS charging pumps.

Exhaust from the vent system is directed to the Pressurizer Relief Tank (PRT) and therefore will not impinge upon vital equipment. The PRT is located in the lower containment which is ventilated and cooled by four air handling units. In addition, the hydrogen skimmer system has ducts in the lower containment high points to disperse any accumulated hydrogen.

Assuming that 100% hydrogen is being vented from the reactor vessel head, the vent system flow rate is 690 cfm. This allows the venting of a gas volume of one-half the RCS in approximately ten minutes.

The power-operated relief valves (PORV) are used to vent the RCS pressurizer. The PORV's are discussed in the McGuire FSAR Section 5.2.2. The RCS vent is located at the top of the reactor vessel head which is the high point of the reactor vessel and coolant loops. This system in conjunction with the PORV's provides a venting capability for the entire RCS with the exception of the U-tube steam generators.

A postulated break of the reactor vessel head vent line upstream of the flow limiting orifice would result in a small LOCA of not greater than one inch diameter. Such a break is similar to those analyzed in WCAP-9600 for hot leg breaks or pressurizer vapor space breaks. Extensive discussions were provided in WCAP-9600 regarding the applicability of the break flow models employed as well as other specific modeling features employed for small LOCA analysis. System response for this postulated break location would closely parallel that described in Section 3.2 of WCAP-9600. Since the break location in the head vent line would behave similarly to the hot leg break case presented as Case F of that section, the discussion presented in WCAP-9600 for that study applies to this postulated break. As such, this postulated break would result in no calculated core uncovery.

The Westinghouse Owners' Group has approved a program to develop appropriate instructions regarding the conditions under which reactor vessel head vent operations should be conducted, and the manner in which head vent operations would be made. This program is scheduled to be completed prior to the end of this year.

It should be noted that the McGuire Nuclear Station already meets the requirements of 10CFR 50.46 and 10CFR 50.44 for design basis accidents. The head vent system and its operation will not compromise this ability. The instructions developed as part of the above referenced Westinghouse Owners' Group effort will be developed specifically to enhance the safe recovery of the plant for events beyond the design base accounting for combustible gas limit considerations and natural circulation.

A flow diagram of the McGuire RCS including the RCS vent system is provided in the McGuire FSAR Figures 5.1-1 and 5.1-2.

INADEQUATE CORE COOLING INSTRUMENTS

References: NUREG-0578 - 2.1.3b
Action Plan - II.F.2

Subcooling Monitor

The margin to saturation will be calculated from Reactor Coolant System (RCS) pressure and temperature measurements (wide-range and low-range pressures, wide-range hot leg temperatures, and temperatures from in-core thermocouples). When RCS pressure is below 800 psig wide-range and low-range pressure inputs are compared, and if the inputs agree within 20 psig the low range pressure inputs are used. The wide range pressure inputs are used for the remaining conditions. The in-core thermocouple readings (65) are averaged and compared with the four wide-range hot leg temperatures (RTD's). The highest of these temperatures and the appropriate pressure are then used to calculate a conservative 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 conservative margin to saturation conditions, that is, a plot of plant pressure and temperature in relation to a computer generated saturation curve. In addition, the following numerical values are displayed: each RCS hot leg temperature, RCS pressure, power level, margin to P_{sat}, each RCS loop margin to test, thermocouple average margin to P_{sat}, and the minimum allowable margins to P_{sat} and T_{sat}. Alarm status is indicated by flashing the alarming parameter on the CRT graphic display, the Alarm CRT, and by printout on the Alarm Typer. Two alarm setpoints are provided for both T_{sat} and P_{sat}. The alarm setpoints are dependent on reactor power. Further details on this subcooling monitor are provided in the table which follows.

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.

Reactor Vessel Level Measurement

Duke will install the Westinghouse designed reactor vessel level measurement system in McGuire Unit 1. This system is designed to monitor directly the water level in the reactor vessel, or the approximate void content under forced circulation conditions, during certain postulated accident conditions. Included is equipment to monitor both the upper plenum (head) level, as well as the entire height of the reactor vessel.

The system instrumentation permits vessel level measurement from the bottom to the top of the reactor vessel, utilizing taps off of an existing spare head penetration and a tap off of a thimble tube at the seal table. Two sets of differential pressure transmitters are provided which have differing measurement ranges to cover different flow behavior with and without pump operations. The narrow range cells indicate water level when zero or one reactor coolant pump is operating. The wide range cells indicate the combined core and internals pressure drop for any combination of operating reactor coolant pumps. The upper plenum measurement is taken by two differential pressure transmitters between the same spare head penetration, and taps off two hot legs.

To minimize containment post-accident environment effects in measurement, accuracy, the system design is based upon locating the transmitters outside the containment. Hydraulic isolators in the impulse lines provide the required double barrier protection between the RCS and outside containment. Reference leg temperature measurements, together with the existing RCS temperature and pressure, are utilized to automatically compensate for difference in coolant and reference leg temperature effects.

The best schedule for delivery of the sensors and transmitters utilized in this system is April, 1981. The design support for installation of this system will be as complete as possible prior to receipt of this equipment. Installation and functional testing will be performed during the first subsequent outage of sufficient duration.

INFORMATION PROVIDED ON THE SUBCOOLING METER

Display

Information Displayed (T-Tsat, Tsat, Press, etc.)	T-Tsat, P-Psat Minimum allowable T-Tsat, Minimum allowable P-Psat Temp., Press % 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 Corresponding minimum allowable P-Psat
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)	99.26% (1979 Average)
Single or redundant calculators	SINGLE
Selection Logic (highest T., lowest press.)	HIGHEST VALID TEMPERATURE 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)	65 IN-CORE T/C; One wide range RTD per loop
Range of temperature sensors	T/C: 0-2300°F RTD: 0-700°F

PLANT SHIELDING

References: NUREG-0578 - 2.1.6b
Action Plan - II.B.2

General Design Criteria (GDC) 19 of Appendix A to 10CFR 50 and 10CFR 20 require control of radiation exposure to personnel associated with nuclear station operations. In addition, GDC 4 of Appendix A to 10CFR 50 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 conducted 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.)

Plant systems or portions of systems which might contain significant levels of radioactivity as a result of a Design Basis Accident were selected for the station accessibility review. Included in the review were:

1) those portions of the Residual Heat Removal, Reactor Building Spray, Safety Injection, and Chemical and Volume Control Systems which could be aligned for recirculationof water from the containment sump to the Reactor Coolant System,

2) those portions of the Liquid Waste Recycle System which would collect and store leakage from the systems mentioned in item no. 1,

3) those portions of the Nuclear Sampling System which would be used in determining radiation levels inside containment or those systems mentioned above, and

4) those portions of the Chemical and Volume Control System which supply seal water to the Reactor Coolant Pump seals, and

5) the Waste Gas System and those portions of the Chemical and Volume Control System which could be used to degas the primary coolant.

To aid in identifying potential personnel access problems, the station was divided into post-LOCA radiation zones. Included in the radiation zones were all areas necessary for personnel access in controlling and mitigating a possible accident. Two types of area access have been designated: 1) continual occupancy and 2) infrequent occupancy.

Areas designated as requiring continual access are: 1) Control Room, 2) On-site Technical Support Center (TSC), 3) Onsite Operational Support Center (OSC), and 4) Personnel Access Portal (PAP). The Control Room, TSC, and OSC are located in the Control Complex. The PAP is located in the Administration Building and serves as the main personnel access point to the station proper. The Control Room serves as the initial onsite center of emergency control and is designed to evaluate, control, and respond to various accident conditions. A detailed descroption of Control Room design and functions is presented in Section 7.8 of the McGuire FSAR.

Areas requiring infrequent access are generally located in Auxiliary Building corridors. Two exceptions to this are: 1) Containment Hydrogen Recombiner controls and 2) Emergency Diesle Generators. Redundant hydrogen recombiners are located in the Upper Containment. Power control panels for these recombiners are located in the Electrical Penetration Rooms. Redundant diesel generators are located in a section of the Auxiliary Building designated as the Diesel Generator Area. Various control functions associated with the diesel generators and supporting systems are located in the Diesel Generator Area. Typical functions centered in Auxiliary Building corridors are: 1) station radioactive waste control panels, 2) motor control centers, and 3) instrumentation panels for various station systems.

The major emphasis of the McGuire Nuclear Station plant shielding review was to assure that station personnel would be able to carry out their emergency procedures. The review featured the consistent use of the defined source terms in conjunction with the KAP-6 computer code. Listed below are all areas where radiation problems may exist and their present status.

1. Sample Room - Unit No. 1

   New sampling panels have been designed to allow analysis of reactor coolant and containment atmosphere samples under accident conditions. However, samples from shared process systems are taken from the Unit 1 sample room. A large cable tray penetration is located in the wall separating the sample room from the Unit 1 penetration area. The locations of the cable tray opening and recirculation piping within the penetration area allow significant radiation steaming into the sample room. Using the radiation sources discussed earlier, the sample room will be inaccessible for slightly more than one week. However, the following samples have been identified as needing to be analyzed before one week:

   1. Recycle Holdup Tank
   2. Waste Evaporator Holdup Tank
   3. Waste Drain Tank
   4. Boron Recycle Evaporator Condensate Demineralizer Outlet
   5. Waste Evaporator Condensate Demineralizer Outlet

   Due to the complexity of the geometries involved, shielding of the Unit 1 sample room is not a viable solution. Modifications which would allow analysis of these samples in the Unit 2 sampling room would not be completed until mid 1982.

   An alternate method for collecting these samples through tell-tale drain lines has been established. A review of tell-tale valve locations has shown that all valves are located in the Auxiliary Building corridors and are therefore accessible. Procedures for collecting these samples will be written prior to January 1, 1981, and no design modifications are necessary.

2. Floor Drain Tank Room

   The RHR sump pumps discharge to the Floor Drain Tank. Manual valves associated with isolating the Floor Drain Tank, and directing its contents elsewhere for storage or processing, are located in this room. As a result of the location and manual operation of these valves, personnel exposures could exceed GDC 19. Reach rods have been added to these valves.

Duke Power Company is currently conducting a through review of the environmental qualifications of Class 1E equipment at McGuire. This review includes possible radiation environments resulting from the source terms assumed in the plant shielding review.

IN-PLANT RADIATION MONITORING

References: NUREG-0578 - 2.1.8c
Action Plan - III.D.3.3 (Partial)

Portable air samplers with silver zeolite radioiodine sampling cartridges are used at McGuire for sampling air when the presence of noble gases is suspected. McGuire Health Physics personnel are knowledgeable in the appropriate station procedures and are trained in the equipment required to determine airborne iodine concentrations in the plant under all conditions.

Two independent counting rooms are available for performing detailed sample analysis. These counting rooms have been designed with shielding to reduce radiation levels to 0.02 mR/hr from plant sources during normal operation. Also included in the counting rooms are shielded GeLi detectors, and shielded sample storage areas. One counting room is located in the Administration Building and one in the Auxiliary Building. In addition, a counting room is available in the Duke Technical Training Center located just outside the McGuire exclusion boundary.

A procedure to determine airborne radioiodine concentrations will be established which does not rely on the availability of a counting room. This procedure will utilize portable "survey-type" instrumentation with energy discrimination for iodine to determine a "go" or "no go" iodine concentration for respiratory equipment use. The use of silver zeolite radioiodine sampling cartridges minimizes the amount of Xenon interference. The "go" or "no go" count rates are dependent upon the calibration of the instrument used. The results of this analysis will be available within ten minutes. This instrumentation in conjunction with the portable air samplers is a fully adequate method to monitor iodine in-plant and meets the November 19, 1979 "clarifications" to NUREG 0578, Item 2.1.8c.

To reduce counting system saturation, sample sizes will be varied to minimize counting system problems. In addition, nitrogen purging of the counting room GeLi detector shields can be used to reduce airborne activity interferences.

DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

August 15, 1980

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

Attention: Mr. B. J. Youngblood, Chief
Licensing Projects Branch No. 1

Re: McGuire Nuclear Station
Units 1 and 2
Docket Nos. 50-369, 50-370

Dear Mr. Denton:

Attached is Duke Power Company's response to the human factor review of the McGuire control room by the NRC Staff. These responses are concise and direct since all of the items have been discussed in detail with the Staff in a meeting on July 30, 1980 in Bethesda. In that meeting and previous meetings, slides and handouts were used to outline in some detail our effort to improve the control room as a result of both the NRC review and Duke's own review. This has involved a four man-year effort on Duke Power Company's part to date. The attached represents what we perceive to be acceptable resolutions for each of the items identified by the NRC review team.

If there are questions regarding the attached, please advise.

Very truly yours,

William O. Parker, Jr.

GAC:vr
Attachment

bcc: K. S. Canady
R. S. Darke
. McMeekin
. Wylie
. Coley
. Harrall
er File MC 801.01
tion File MC 801.01

NRC REQUEST FOR INFORMATION TRANSMITTED
BY LETTER OF JUNE 4, 1980 FROM B. J. YOUNGBLOOD,
CHIEF, LICENSING BRANCH NO. 1
DIVISION OF LICENSING

1. Provide additional information and evaluations for the effect of high post-accident radiation levels on access and equipment operation in vital areas as follows:

   (a) Identify those areas considered vital areas for post-accident recovery, considering the following areas: control room, on-site technical control center, operational control center, security center, emergency power supplies, radioactive waste control panels, recombiner hookup and controls, hydrogen purge controls, instrument panels, containment isolation valve reset controls, sampling stations, sample analysis stations, manual ECCS alignment, motor control center. State which areas will not require access under postulated post-accident conditions and explain why any particular area is not considered a vital area for the post-accident criteria referenced.

   (b) Identify those systems which may contain high levels of radioactivity and have been evaluated for effects on access to and operations in vital areas. Consider (as a minimum) residual heat removal safety injection, CVCS, demineralizer, charging, RC filters, seal water filter, liquid radwaste, gaseous radwaste. Provide an explanation for any of the above systems not evaluated or considered.

   Response

   See Plant Shielding

2. Revise and broaden your response of January 24, 1980 so as to provide a description of the two high range containment monitors required by our letter of November 9, 1979, implementing the Lessons Learned item 2.1.8.b of NUREG-0578, and specify the location of these monitors (inside containment). The description of the monitors should include:

   a. type of radiation measured
   b. the range or ranges of the monitors. If two or more monitors are required to span the range in Table 2.8.1.b.3 of our November 9, 1979 letter (10^8 rad/hr total radiation or 10^7 R/hr photons only), the ranges of the subsystem monitors must overlap (i.e., upper value/low value of overlap) by at least a factor of 10;
   c. location of and type of readout (continuous and recording);
   d. energy response (sensitive to 60 kev);
   e. calibration frequency and methods (refueling frequency);
   f. verification that the monitors are powered by separate vital instrument buses;
   g. verification that the monitors will be operational by January 1, 1981;
   h. verification that the monitors meet the seismic qualifications of Regulatory Guide 1.100 (Seismic Category I) and are environmentally qualified to survive an in-containment LOCA in accordance with Regulatory Guide 1.89.

The location of the monitors should be shown on plant layout drawings. The monitors should be located in a manner as to provide a reasonable assessment of radiation levels inside containment. Monitors should not be placed in areas which are protected by massive shielding.

Response

See Additional Accident Monitoring Instrumentation

3. Describe the provisions which have been made to sample and analyze airborne radioiodine in vital areas as follows:

   a. portable monitoring capability for radioiodine sampling;
   b. results available within 10 minutes at a time when usage of counting analyses facilities is heavy;
   c. controls to prevent counting system saturation from high sample activities;
   d. clean air ventilation for counting facilities to reduce analysis inaccuracies;
   e. provisions for reducing background radiation levels in the analysis facility, in the sample counting device, or from the sample;
   f. procedures for keeping personnel exposures ALARA during sample taking and analysis.

Response

See In-Plant Radiation Monitoring

4. Your FSAR indicates that the Health Physics Supervisor may have direct access to the Station Manager in matters concerning any phase of radiation protection (p. 13.1-13), the station organization chart in Figure 13.1.2-1 shows Health Physics Supervisor reporting through the Technical Services Superintendent to the Plant Manager. Regulatory Guide 8.8, "Information Relevant to Ensuring Occupational Radiation Exposures at Nuclear Power Stations will be As Low As is Reasonably Achievable," states that the Radiation Protection Manager (RPM), equivalent to your Health Physics Supervisor, should have direct recourse to the plant manager and should be independent of the station technical support division. The "Draft Criteria for Utility Management and Technical Competence" specifies that the RPM should report directly to the Plant Manager. It is our position that your Health Physics Supervisor report directly to the Station Manager and that the Station organization be revised accordingly.

Response

The McGuire Station Health Physicist reports directly to the Superintendent of Technical Services. He does have direct access to the Station Manager regarding matters of radiological protection. This direct access is documented in the McGuire FSAR, the McGuire Station Health Physics Manual, and the McGuire Station Health Physicist position guide. The McGuire Station Manager has the final responsibility for the protection of persons against radiation.

On June 27, 1980 a NRC management review team concluded an audit of Duke Power Company's readiness to operate McGuire Nuclear Station. This team utilized the "Draft Criteria for Utility Management and Technical Competence" as one of the bases for their review. The team reviewed the McGuire station organization and identified no deficiencies regarding the Health Physics organization.

Duke believes that the existing McGuire station organization provides for adequate protection of the health and safety of the public and plant personnel. Therefore, no changes in the station organization are planned.

NRC REQUEST FOR INFORMATION TRANSMITTED
BY LETTER OF JULY 23, 1980
FROM R. L. TEDESCO, ASSISTANT DIRECTOR FOR LICENSING
DIVISION OF LICENSING

1. Provide your design of the pressurizer vent system per our November 9, 1979 letter. Assure that all requirements and recommendations of that letter are satisfied.

2. Demonstrate that the vent system is qualified to RPS safety grade standards. Include seismic design, IEEE-279 requirements, minimize inadvertent actuations, vent valve position indication in the control room, qualification to pass noncondensibles, steam, water and combinations thereof, etc.

3. Demonstrate that the exhaust from the vent system will not impinge on other equipment, that the vent system will vent RCS hot legs, and that the vent exhaust is to a portion of the containment with maximum ventilation and cooling.

4. Provide an analysis of the vent system discharge capacity Assure that the 1/2 RCS volume per hour guideline of our November 9, 1979 letter is satisfied.

5. Provide justification that the thermal-hydraulic characteristics for a break in the vent line at the reactor vessel head is adequately modeled. This justification should consider the differences between the break location of the vent and the classical LOCA analyses. It should consider the differences associated with flow distribution and depressurization rate.

6. Provide procedural guidelines and analytical bases (preferably, generically developed by owners group) for vent operation and termination as related to plant performance. The procedures should be based on the following criteria: (1) the plant must meet the requirements of 10CFR 50.46 and 10CFR 50.44 for DBA's; and (2) the plants ability to maintain core cooling and containment integrity for events beyond DBA's must be increased. Procedures should also address methods to (1) assure natural circulation through the U-tube portions of the steam generator with the potential accumulation of gases in this region, and (2) assure that combustible limits are not exceeded.

Response

See Reactor Coolant System Vents

09/08/80

NRC REQUEST FOR INFORMATION TRANSMITTED
BY LETTER OF AUGUST 25, 1980 FROM R. L. TEDESCO,
ASSISTANT DIRECTOR FOR LICENSING, DIVISION OF LICENSING

1. The McGuire response to NUREG-0578, Item 2.1.8.c proposes the use of portable survey instruments in conjunction with silver zeolite cartridges to determine radioiodine concentrations for respiratory use. Explain whether this procedure will be used as a substitute for, or in conjunction with standard radioiodine sampling and analysis systems outlined in the "clarifications" of the NUREG-0578 clarification letter dated November 19, 1979. Also provide a description of the method for evaluation.

Response:

See In-Plant Radiation Monitoring

2. Provide a commitment to complete alternate auxiliary building telltale valve sampling procedures by January 1, 1981 in lieu of modifications to the Unit 1 sampling room.

Response:

See Plant Shielding

3. Provide an explanation as to why the Waste Gas Treatment System was not included in the evaluation as a potential highly radioactive system or provide the results of the evaluation of this system.

Response:

See Plant Shielding

4. Verify that the environmental and seismic qualifications of the proposed radiation monitors meets the position of NUREG-0578, Item 2.1.8.b as stipulated in the requirements of Regulatory Guides 1.89, 1.97 Rev. 2 (ANSI N320-1978) and 1.100.

Response:

The response to this request will be provided at a later date.

5. Provide a commitment to calibrate the proposed high radiation monitors at each refueling outage (at least every 18 months) as a minimum in accordance with the staff position on NUREG-0578, Item 2.1.8.b and according to the Technical Specifications for all Radiation Monitoring Instruments (3/4 3.3, Table 4.3-3).

Response:

The response to this request will be provided at a later date.