MISSISSIPPI POWER & LIGHT COMPANY
Helping Build Mississippi
P. O. BOX 1640, JACKSON, MISSISSIPPI 39205

October 23, 1981

NUCLEAR PRODUCTION DEPARTMENT

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Attention: Mr. Harold R. Denton, Director

Dear Mr. Denton:

SUBJECT: Grand Gulf Nuclear Station
Units 1 and 2
Docket No. 50-416 and 50-417
File 0260/L334.0/L350.0
Response to SER Items
1.10(22), 1.11(17), and
1.9(8)
AECM-81/410

In accordance with your request for additional information in support of the Grand Gulf Nuclear Station Safety Evaluation Report, NUREG-0831 (SER), Mississippi Power & Light Company is providing the enclosed information pertaining to Post-Accident Sampling (attachment 1), Fire Protection (attachment 3), Containment Isolation (attachment 5), and RHR/Containment Spray Cooling (attachment 6); SER items 1.11(17), 1.10(22), 1.9(8) and 1.10(15) respectively.

Additionally provided are responses to concerns directed to Mississippi Power & Light Company during a meeting between members of our staff and the Chemical Engineering and Radiological Assessment Branches on August 19, 1981. This information is presented in Attachment 2.

Portions of the enclosed information represents changes to the Grand Gulf Safety Analysis Report (FSAR). Appropriate changes will be made in a forthcoming amendment to the FSAR. If you have any questions or require additional information, please contact this office.

Yours truly,

L. F. Dale
Manager of Nuclear Services

RFP/JGC/JDR:pn

Attachments: (See Next Page)

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S1/1

Attachments:
1. Question and Response 281.9
2. Concerns and responses resulting from a meeting between members of our staff and the Chemical Engineering and Radiological Assessment Branches on August 19, 1981
3. Fire Protection: Comparison of the GGNS Fire Protection Program to 10CFR50, Appendix R
4. Draft procedure "Core Damage Estimation"
5. Containment Isolation, CSB concern
6. RHR/containment spray cooling

cc: Mr. N. L. Stampley
Mr. R. B. McGehee
Mr. T. B. Conner
Mr. G. B. Taylor

Mr. Victor Stello, Jr., Direactor
Office of Inspection & Enforcement
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

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281.9 (TMI II.B.3) Provide information that satisfies the attached proposed license conditions for post-accident sampling.

Attachment to 281.9:

NUREG-0737, II.B.3 - Post Accident Sampling Capability

REQUIREMENT

Provide a capability to obtain and quantitatively analyze reactor coolant and containment atmosphere samples, without radiation exposure to any individual exceeding 5 rem to the whole body or 75 rem to the extremities (GDC-19) during and following an accident in which there is core degredation. Materials to be analyzed and quantified include certain radionuclides that are indicators of severity of core damage (e.g., noble gases, iodines, cesiums and non volatile isotopes), hydrogen in the containment atmosphere and total dissolved gases or hydrogen, boron and chloride in reactor coolant samples in accordance with the requirements of NUREG-0737.

To satisfy the requirements, the application should (1) review and modify his sampling, chemical analysis and radionuclide determination capabilities as necessary to comply with NUREG-0737, II.B.3, (2) provide the staff with information pertaining to system design, analytical capabilities and procedures in sufficient detail to demonstrate that the requirements have been met.

EVALUATION AND FINDINGS

The applicant has committed to a post-accident sampling system that meets the requirements of NUREG-0737, Item II.B.3 in Amendment 49, but has not provided the technical information required by NUREG-0737 for our evaluation. Implementation of the requirement is not necessary prior to low power operation because only small quantities of radionuclide inventory will exist in the reactor coolant system and therefore will not affect the health and safety of the public. Prior to exceeding 5% power operation the applicant must demonstrate the capability to promptly obtain reactor coolant samples in the event of an accident in which there is core damage consistent with the conditions stated below.

1. Demonstrate compliance with all requirements of NUREG-0737, II.B.3, for sampling, chemical and radionuclide analysis capability, under accident conditions.

2. Provide sufficient shielding to meet the requirements of GDC-19, assuming Reg. Guide 1.3 source terms.

3. Commit to meet the sampling and analysis requirements of Reg. Guide 1.97, Rev. 2.

4. Verify that all electrically powered components associated with post accident sampling are capable of being supplied with power and operated, within thirty minutes of an accident in which there is core degradation, assuming loss of off site power.

5. Verify that valves which are not accessible for repair after an accident are environmentally qualified for the conditions in which they must operate.

6. Provide a procedure for relating radionuclide gaseous and ionic species to estimated core damage.

7. State the design or operational provisions to prevent high pressure carrier gas from entering the reactor coolant system from on line gas analysis equipment, if it is used.

8. Provide a method for verifying that reactor coolant dissolved oxygen is at 0.1 ppm if reactor coolant chlorides are determined to be 0.15 ppm.

9. Provide information on (a) testing frequency and type of testing to ensure long term operability of the post accident sampling system and (b) operator training requirements for post-accident sampling.

10. Demonstrate that the reactor coolant system and suppression chamber sample locations are representative of core conditions.

In addition to the above licensing conditions the staff is conducting a generic review of accuracy and sensitivity for analytical procedures and on-line instrumentation to be used for post-accident analysis. We will require that the applicant submit data supporting the applicability of each selected analytical chemistry procedure or on-line instrument along with documentation demonstrating compliance with the licensing conditions four months prior to exceeding 5% power operation, but review and approval of these procedures will not be a condition for full power operation. In the event our generic review determines a specific procedure is unacceptable, we will require the applicant to make modifications as determined by our generic review.

RESPONSE

Condition 1: Compliance with the requirements of NUREG-0737, Item II.B.3, "Postaccident Sampling Capability", is demonstrated by the information given in the following subsections:

NUREG-0737, Item II.B.3	Subsection
1	7.7.1.11.4.2.a
2	7.7.1.11.4.2.b
3	7.7.1.11.4.2.c
4	9.3.2.2.4
5	7.7.1.11.4.2.d
6	7.7.1.11.4.2.e
7	*Grab sample only-see Table 9.3-3

NUREG-0737, Item II.B.3	Subsection
8	7.7.1.11.4.2.f
9	7.7.1.11.4.2.g,h,i
10	7.7.1.11.4.2.j
11	7.7.1.11.4.2.k,9.3.2.2.4

*The post accident sampling system does not have the capability to analyze for boron concentration with in-line equipment. The post accident sampling system is capable of providing a post-accident grab sample which can be analyzed for boron concentration onsite (if radiation levels permit) or can be sent offsite for boron concentration analysis.

Condition 2: Sufficient shielding is provided to make it possible for an operator to obtain and analyze a sample without radiation exposures exceeding the criteria of GDC 19 assuming Regulatory Guide 1.3 source terms. See subsections 7.7.1.11.4.2.e and g.

Condition 3: The sampling and analysis requirements identified in Regulatory Guide 1.97 Revision 2 under Accident Sampling Capability will be met with one exception. Grab sample provisions for sumps is not deemed necessary due to the design of the Mark III containment and the system capability of allowing on line and grab sample analysis of the suppression pool.

Condition 4: The post accident sample system components required for drawing and analyzing a sample of the reactor coolant and the drywell and containment atmosphere are powered from normal as off-site power. However, in response to the above NRC concern, a method will be provided to restore these components to available on-site power after an accident when off-site power has been lost.

Details of the proposed modifications will be provided by December 31, 1981. Implementation of proposed modifications will be completed prior to startup after the first refueling outage.

Condition 5: The only valves which are inaccessible for repair after an accident are the inboard containment isolation valves (see Table 6.2-44). All these valves are fully qualified (including environmentally) for operation as containment isolation valves.

Condition 6: A method for relating radionuclide gaseous and ionic species to estimate core damage is being devised by General Electric for the fuel type being used at Grand Gulf. Completion of this package is presently scheduled for May 1982 with an interim report available August 1982. Until receipt of this package, a temporary failed fuel estimation procedure will be prepared on the basis of I-131 concentrations as identified in the McGuire Nuclear Station procedure on the same subject. A draft of this interim procedure is enclosed (attachment 4) for your review.

Condition 7: Carrier gas is not used in the Grand Gulf Post Accident Sampling System.

Condition 8: As stated in subsection 7.7.1.11.4.2.j, the post accident sampling system utilizes instrumentation with ranges adequate to provide pertinent data to the operator. In addition, the ranges of the instruments at least meet those ranges identified in Regulatory Guide 1.97, Revision 2.

Condition 9: The Post-Accident Sampling System will be used to perform at least monthly reactor coolant sample analyses for gamma isotopic, chloride, conductivity, pH, oxygen and hydrogen. Every six months, for training and operability testing, a diluted liquid grab sample will be drawn, transported, and analysed in the Hot Lab for boron. This sample will be handled as a post accident highly radioactive sample. In addition, every six months, a containment air sample will be analyzed for hydrogen, oxygen and gamma isotopic. Classroom training will also be provided on system operation and the proper handling of highly radioactive samples.

Condition 10. As shown in Table 9.3-3, the post accident sample system is also capable of drawing and analyzing samples of the containment and drywell atmosphere. The locations of these sample lines, as shown in Figure 7.5-5, are in relatively open areas of containment with adequate communication.

The non-condensable hydrogen and fission product noble gases that will be released to the drywell and containment post accident are assumed to form a homogeneous mixture. The even mixing of non-condensable gases will be promoted by:

a. Natural convection as a result of temperature gradients in the drywell and the cascading effect of ECCS water exiting from a break.
b. Turbulence resulting from the operation of containment sprays.
c. Depressurization of the reactor coolant system via the sequential opening of safety-relief valves distributed around the suppression pool. This will result in an approximate uniform distribution of non-condensables in the containment.
d. Turbulence resulting from the localized burning of hydrogen as initiated by the Hydrogen Ignitor System.

Based on the above, the drywell and containment atmosphere samples will be representative of actual conditions.

A representative core sample is directly dependent on the amount of mixing of the reactor coolant from the core region with that of the sample location. Obtaining a sample of reactor coolant that is representative of core conditions is achieved normally by sampling from the recirculation system via the recirculation loop B sample point as listed in Table 9.3-1. Therefore, adequate mixing is accomplished by forced circulation provided by the recirculation pumps. However, core flow circulation in the BWR is inherent without the use of recirculation pumps. Lack of communication between the downcomer and the core could result in disruption of the major natural circulation flow. This situation is not likely to occur in jet pump plants because of the open communication between the regions. In cases where the recirculation pumps are not available, naturally induced coolant flow is established or can be maintained by the density difference between the downcomer region and the core provided such density difference head is sufficient to balance the losses in the loop.

The primary natural circulation loop is between the downcomer and the core (See Figure 281.9-1). Due to boiling in the core region, a large difference in densities is available for driving natural circulation flow from the downcomer through the jet pumps and into the shroud region. The flow due to natural circulation is given in Figure 4.4-5.

The circulation flow of primary coolant by natural circulation is sufficient to provide mixing of the primary coolant such that a sample taken from the jet pump diffuser location would be representative of core conditions.

Obtaining a representative sample therefore is dependent upon maintaining natural circulation flow. First, for a accident such as a DBA LOCA, it is assumed that the majority of the flow of reactor coolant will be the path of least resistance and will be out of the break area eventually into the suppression pool with inventory being maintained by ECCS systems. For this worst case then, sufficient mixing would still take place in that effectively the entire suppression pool and reactor coolant become the same fluid and a sample from the suppressional pool or the jet pump would be representative.

The Grand Gulf analysis indicates that the absolute minimum water level for natural circulation between the downcomer and the core region is the elevation of the jet pump suction inlet, which is 8.34 ft above the elevation of the BAF for the GGNS vessel and about 21 ft below normal water level. An analysis of small breaker accidents (SBA) with all systems operable was provided to the NRC in NEDO 24708A Revision 1, December, 1980. As can be seen from that analysis, the water level both inside and outside the core shroud did not fall below the top of the active fuel (TAF). It is assumed therefore that for a SBA, sufficient inventory is conservatively maintained such that

water level is maintained ≥ 8.34 ft above the bottom of the active fuel (BAF) or at the level of the jet pump suction. For water levels higher than the jet pump suction inlet, naturally induced flow will be maintained by the density difference between the downcomer region and the core provided such density difference head is sufficient to balance the irreversible losses in the Loop. The minimum required water level can thus be higher than the jet pump inlet and will depend on the pressure in the RPV and decay heat. The latter is a function of time after scram. This level is thus determined for chosen RPV conditions by balancing the hydrostatic driving head between the downcomer and the core against the pressure drop for vanishing flow thus different RPV levels were considered with decay heat generation spanning a time period up to several days after shutdown. The minimum required downcomer water levels in order to provide natural circulation through the core are summarized below:

Minimum Downcomer Water Level (Ft above BAF) for RPV Internal Natural Circulation

		RPV Pressure (PSIA)		
Time from Scram	Decay Heat* (% rated core power)	1035	500	15
20 seconds	4.3	13	9	8.3**
4 hours	0.86	23	16	8.3**
7 days	0.086	33	24	12.0

* ANS Standard 5.1 September 1978 revision
** Elevation of Jet Pump Suction Inlet

It will be noted that for a decay heat level less than 0.1% at a system pressure of 1035 psia the required level (33 ft)+ is well above the normal operating water level (29.5 ft)+. It is however, unlikely that such a high system pressure will exist seven days from scramming the control rods and the calculation is intended as a reference only.

+ Reference level is Bottom Active Fuel

The Grand Gulf analysis indicates that natural circulation will continue to function as the reactor core is cooled. Core flow calculations as a function of coolant temperature were carried out at an RPV pressure of 15 psia and at 1% decay heat rate (2.3 hrs. from scram). The results are as follows:

Coolant Temp.°F	Natural Circulation as a % of Nominal Core Flow
200	1.9
190	1.3
180	1.1

At a decay power level of .1%, equivalent to about 7 days from a scram and at a coolant temperature of 200°F, natural circulation will still continue but the rate will be 0.2% of nominal core flow.

Based on the Grand Gulf analysis, we are confident that provisions exist for adequate mixing of the core and downcomer fluids and that representative samples are obtainable from the sample points indicated for reactor coolant in Table 9.3-3.

Additionally, the sample taken monthly from the Post Accident Sample System (PASS) will be compared to the normal (Recirc. Loop B) sample to ensure that meaningful results are obtained via this sample path.

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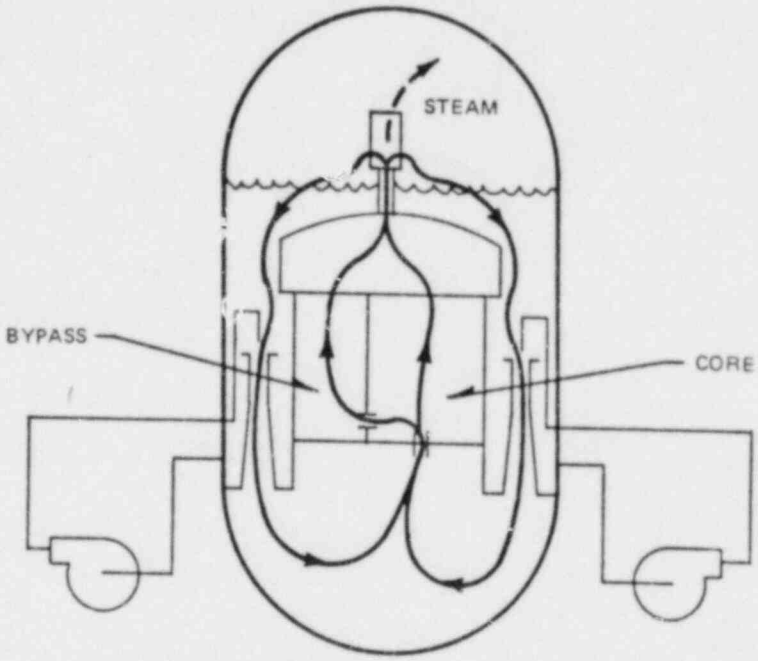

NORMAL NATURAL CIRCULATION

Figure 281.9-1

The following concerns were directed to MP&L during a meeting between members of our staff and the Chemical Engineering and Radiological Assessment Branches on August 19, 1981. The appropriate information from these responses will be incorporated into the next available FSAR amendment.

CONCERNS:

1. What other plants use the TEC Post Accident Sample System?

RESPONSE

To date, TEC has not sold any utility their Post Accident Sample System package.

2. Provide a copy of GE document 22A5718, "Mark III Containment Dose Reduction Study", to John Minns NRC-RAB.

RESPONSE

GE document 22A5718 was referenced in response to NRC Question 331.29. A copy of 22A5718 was provided informally to John Minns the week of August 17, 1981.

3. Verify that all electrically powered components associated with post accident sampling are capable of being supplied with power and operated, within thirty minutes of an accident in which there is core degradation, assuming loss of off site power.

RESPONSE

The response to this action item is given in response to NRC Question 281.9, part 4.

4. Provide the procedure, time to complete, shielding provided and the expected dose rate regarding the transfer of undiluted samples for offsite analysis.

RESPONSE

Samples for offsite analysis:

To obtain a sample, it is estimated that the operator will spend 1 to 2 minutes in the sample area connecting the two quick-connect fittings prior to sampling, closing the two cask valves, opening the bypass valve after sampling, and uncoupling the quick-connect fittings after purging the lines. During the remainder of the time, the operator is outside the sample room in the corridor of the Turbine Building at the sample control panel.

After the sample has been collected, the sample cask will be moved from the sample room on the 93 foot elevation of the Turbine Building north along the corridor to an equipment hatch or the same elevation at the north end of the Turbine Building. The sample (10 ml of $\approx$ 10Ci/cc reacator coolant) will be contained in a 1200 lb shielded sample cask

(which is carried on a motorized sample cart). The sample cask design provides for a dose level of ≤ 100 mR/hr at the surface of the sample cask.

The sample cask will then be attached to the Turbine Building crane or other suitable lifting device and lifted to Turbine Building elevation 133 feet. The sample cask will then be moved from the equipment hatch directly to a truck at the loading dock located directly adjacent to the equipment hatch on the 133 foot elevation of the Turbine Building. It is assumed that it will take approximately 10 minutes to transfer the sample cask from the sample station to the truck for transit to the offsite analysis facility.

The sample panel, as well as the sample cask, is lead shielded to limit the operator dose rate to ≤ 100 mR/hr. The motorized cart is approximately 83" long. The operator controls are on one end and the 28½" long sample cask sits on the other end. This provides approximately 4 feet of physical separation between the operator and the sample cask during transport.

5. Indicate the type of connection used on high pressure lines in the post accident sample panel.

RESPONSE

Both Swagelok and welded connections are used for high pressures lines in the sample panel.

6. Provide a description of the 1000:1 dilution chamber used in the post accident sample system.

RESPONSE

A 2.25 ml sample chamber is contained between two three-way motor operated valves. The coolant normally flows through this chamber. When a sample is taken, both three-way valves change position, blocking coolant flow and trapping a 2.25 ml sample. Demineralized water from a 2250 ml chamber is allowed to flow through the sample chamber and into a large mixing chamber. After the sample and Demineralized water have been mixed, the diluted sample flows into the grab sample cylinder.

7. Demonstrate that the reactor coolant system and suppression chamber sample locations are representative of core conditions.

RESPONSE

The response to this action item is given in the response to NRC Question 281.9, part 10.

8. Discuss the accuracy/sensitivity of on-line instrumentation which will be used for post accident sample analysis.

RESPONSE

The response to this action item is given in the response to NRC Question 281.9.

9. Provide the responses to NRC Questions 281.8 and 281.9.

RESPONSE

The response to Question 281.8 was submitted to NRC in MP&L letter, AECM-81/322, dated August 25, 1981. The response to Question 281.9 is provided in Attachment 1 to this letter.

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Table 9A-4

Fire Protection Program - Comparison to 10CFR 50, Appendix "R"

On October 27, 1980 the Commission approved a rule concerning fire protection. The rule and its Appendix R were developed to establish the minimum acceptable fire protection requirements necessary to resolve certain areas of concern in contrast between the NRC staff and licensees of plants operating prior to January 1, 1979.

This fire protection rule does not apply to Grand Gulf Nuclear Station; however, as a result of a meeting held with the NRC staff on June 30, 1981 and at the staff's request, a comparison of the Grand Gulf Nuclear Station fire protection program to the requirements outlined by 10CFR 50 Appendix R, Section II and III is given below:

Appendix R Requirement	Grand Gulf Nuclear Station position/discussion
II GENERAL REQUIREMENTS	
A. Fire Protection Program	Comply. Details of the program are given in Appendix 9B
B. Fire Hazards Analysis	The Grand Gulf fire hazards analysis is described in detail in Appendix 9A, Section 7.0, and summarized in Table 9A-2, the fire hazards analysis included the identification of potential in situ and transient fire hazards, and the determination of the consequences of a fire in any location on the ability to safely shut down the plant. Where necessary, specific fire protection measures were provided to ensure that safe shut down capability was maintained in the event that a postulated fire were to occur.
C. Fire Prevention Features	1. Comply. As discussed in Appendix 9A, Section 7.2, all in situ fire hazards have been identified and suitable fire protection measures provided.

2. Comply. Details are provided in Section 9B.4. Maintain Fire Protection.

3. Comply. As described in Appendix 9A, Section 7.2, and shown on Figures 9A-16 through 9A-33 and 9A-36 through 9A-51, fire detection systems, portable extinguishers, and standpipe and hose stations are installed in strategic locations throughout the plant.

4. Comply. As described in Appendix 9A, Section 7.2, and as shown on Figures 9A-3 through 9A-9, fire barriers and automatic fire suppression systems have been installed in the plant where required to protect redundant safe shutdown-related systems and components.

5. See discussion of item III.H and III.I.

6. Comply. Fire detection and suppression systems have been designed by the Architect Engineer and approved for use by American Nuclear Insurers (ANI). Installation of the systems was performed by trade craftsmen. Maintenance and testing is performed in accordance with approved maintenance and surveillence procedures and the Grand Gulf Technical Specifications under the supervision of personnel properly qualified by experience and training in fire protection systems.

7. Comply. Surveillence procedures have been established and are performed in accordance with the requirements of the Grand Gulf Technical Specifications and Grand Gulf Operations Manual.

D. Alternative or dedicated shutdown capability	As discussed in Appendix 9A, Section 7.2, suitable fire protection measures have been provided to ensure that a fire in any area of the plant will not affect safe shutdown capability. Therefore, alternate or dedicated shutdown capability for a specific area is neither necessary or required. As discussed in Appendix 9A, subsection 7.2.2.46, an exposure fire in the control room which disables both divisions of redundant systems is not considered a credible event. However, in response to an NRC request, electrical isolation will be provided between the control room and the Division I remote shutdown panel prior to startup after the first regularly scheduled refueling outage. See the response to Question 013.18.
III. SPECIFIC REQUIREMENTS	
A. Water Supplies for fire suppression systems	Comply. As described in subsection 9.5.1.2.1, the Grand Gulf fire protection water supply system consists of two 300,000 gallon nominal capacity water storage tanks at atmospheric pressure and three 1500 gpm fire pumps (1 electric, 2 diesel). Each of the three fire pumps has the capability to take suction from either water storage tank. Therefore, an adequate fire water source is constantly available.
B. Sectional Isolation Valves	Comply. As described in subsection 9.5.1.2.2.1, and shown on Figures 9.5-2 and 9.5-3, post indicator valves are provided to permit isolation of portions of the 12" underground fire main loop for maintenance or repair without interrupting the entire water supply.

C.	Hydrant Isolation Valves	Comply. As shown on Figures 9.5-2, 9.5-3 and 9.5-8, valves are installed to permit isolation of outside hydrants from the fire main for maintenance or repair without interrupting the water supply to automatic or manual fire suppression systems in any area containing or presenting a fire hazard to safety-related or safe shutdown equipment.
D.	Manual Fire Suppression	Comply. As discussed in subsection 9.5.1.2.2.2 and shown on Figures 9.5-2, 9.5-3, 9.5-7 and 9.5-8, standpipes and hose streams are strategically located throughout the plant. All areas containing safety-related or safe shutdown equipment are designed to permit effective functioning by the plant fire brigade. All fire suppression systems located inside containment are supplied by the condensate and refueling water storage and transfer system (CRWST) with backup supply available from the fire water loop. The capacity, adequacy, and reliability of the CRWST system is described in the response to Question 013.31. Hose stations located inside containment are provided with sufficient lengths of hose to reach any location inside the drywell with an effective hose stream.
E.	Hydrostatic Hose Tests	Comply. All fire hose shall be tested annually to a pressure of 300 psi.
F.	Automatic Fire Detection	Comply. As described in subsection 9.5.1.2.2.7 and shown on Figures 9A-16, 9A-17, 9A-22, and 9A-28 through 9A-33, automatic fire and smoke detectors are installed in all areas of the plant that contain or present a potential exposure

fire hazard detrimental to safe shutdown or to the operation of safety-related systems or equipment.

All control functions associated with the fire and smoke detection systems are powered from an inverter which is fed from one of the station's BOP batteries. Upon loss of offsite power, battery chargers can be manually connected to one of the standby diesel generators, thereby providing a continued DC power supply for a time period in excess of the normal two hour battery capacity. For a further discussion of the fire and smoke detection system power supply, see subsection 9.5.1.2.2.13.

G. Fire Protection of Safe Shutdown Capability

1. Comply. Active and passive fire protection measures have been provided to ensure hot and cold shutdown capability.

2. Comply. As described in Appendix 9A, Section 7.2, automatic smoke detection and fire suppression systems and one-hour rated fire barriers have been provided outside of containment where three-hour rated fire barriers were not adequate to ensure that a single exposure fire could not affect redundant safe shutdown-related components.

Inside containment, redundant safe shutdown-related components are separated by at least 45 feet, as described in Appendix 9A, subsection 7.3.2.1. Therefore, separation provides adequate protection against the effects of an exposure fire.

3. Comply. Since the requirements for Sections G.1 and G.2 have been satisfied, alternate or dedicated shutdown capability is not required for any area in the plant.

		As discussed in Appendix 9A, subsection 7.2.2.46, an exposure fire in the control room which disables both divisions of redundant systems is not considered a credible event. However, in response to an NRC request, electrical isolation will be provided between the control room and the Division I remote shutdown panel prior to startup after the first regularly scheduled refueling outage. See the response to Question 013.18.
H.	Fire Brigade	Comply. The fire brigade will be staffed and equipped in accordance with the provisions stated. Further details are provided in the responses to questions 13.16, 13.17, 422.18, 422.19, 441.1, and letter dated August 27, 1981 (AECM-81/331).
I.	Fire Brigade Training	Comply. Additional information is available in Section 13.2.4 and letter dated August 27, 1981 (AECM-81/331).
J.	Emergency Lighting	Comply. As discussed in subsection 9.5.3.1.1 and Table 9A-2, Section D.5.a., eight hour emergency lighting has been provided in the control room, remote shutdown panel areas, and in the access and agress routes there to.
K.	Administrative Controls	Comply. Additional information is provided in Appendix 9B Section 9B.4.2.
L.	Alternate or Dedicated Shutdown Capability	As discussed in Appendix 9A, Section 7.2, a fire in any area of the plant will not affect safe

		shutdown capability. Therefore, alternate or dedicated shutdown capability for a specific area is neither necessary or required. As discussed in Appendix 9A, subsection 7.2.2.46, an exposure fire in the control room which disables both divisions of redundant systems is not considered a credible event. However, in response to an NRC request, electrical isolation will be provided between the control room and the Division I remote shutdown panel prior to startup after the first regularly scheduled refueling outage. See the response to Question 013.18.
M.	Fire Barrier Cable Penetration Seal Qualification	Comply. As discussed in subsection 9.5.1.2.2 and Table 9A-1, Section D.3, fire barrier cable penetration seals are qualified and tested in accordance with NFPA, ANI and IEEE standards. The fire barrier penetration rating equals the fire rating of the respective barrier. Additional information on this issue was provided to the NRC at the NRC Staff's request, by letter dated August 21, 1981 (AECM-81/309).
N.	Fire Doors	Comply. Fire doors are provided with self closing mechanisms. Fire doors, when used as security doors, are kept closed and electrically supervised. Other fire doors are kept locked and are periodically inspected to verify that the doors are in a closed position. The fire brigade leader has ready access to keys for any locked fire doors.

O.	Oil Collection System For Reactor Coolant Pump	An exposure fire due to the ignition of the recirculation pump lubricating oil is not a credible event. As described in Appendix 9A, subsection 7.2.3.2, and the response to Question 013.23, the heavy construction and non-pressurized design of the recirculation pumps lubricating system minimizes the susceptability of the system to leakage. The design of the recirculation pumps minimizes piping connections to atmospheric vents, drains, and fill connections. Therefore, an engineered oil containment and collection system is not required.

Volume 08

Section 03

08-S-03-17

Revision 0

Date:

(DRAFT)

CHEMISTRY PROCEDURE

CORE DAMAGE ESTIMATION

SAFETY-RELATED

Prepared: R. Brown

Reviewed: _____ / _____
Technical Review Plt. Quality Superintendent

PSRC: _____

Approved: _____
Chemistry/Radiation Control Superintendent

Concurrence: _____
Assistant Plant Manager

List of Effective Pages:

Page:	Revision
I-3	Rev. 0
Atts. I-V	Rev. 0

1.0 PURPOSE

To estimate the percent failed fuel following a major Reactor Accident from analytical results obtained by sampling the Containment Atmosphere, Suppression Pool and/or Reactor Coolant System.

2.0 RESPONSIBILITY

2.1 The Plant Chemist will provide the Shift Superintendent or Emergency Director with the possible percent fuel damage as soon as the information is available and verified.

2.2 The Lead Laboratory Chemist will perform the initial calculations and provide the attached data calculation sheets and base information to the Plant Chemist as soon as available.

2.3 The most senior laboratory chemist on site will perform the Iodine analysis with another qualified chemist acting as monitor.

3.0 REFERENCES

3.1 NEDO - 20566 - 7, June 1978, General Electric Co., Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, Amend. No. 7

3.2 Duke Power Company Procedure AP/0/A/A5500 "Estimate of Failed Fuel Based on I-131 Concentration"

3.3 GENERAL Physics Corporation - Training Session "Mitigating Reactor Core Damage - Grand Gulf"

3.4 NSAC/14 - Nov. 1980, "Workshop on Iodine Releases in Reactor Accidents"

3.5 USNRC Regulatory Guide 1.97 - 1980, Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident

4.0 ATTACHMENTS

4.1 Iodine-131 Inventory Power Correction (Y)

4.2 Density Correction Factor (X)

4.3 Hydrogen Data Calculation Sheet

4.4 Iodine Data Calculation Sheet

4.5 Core Damage Graph

5.0 DEFINITIONS

5.1 GGNS BASE INFORMATION -

5.1.1 3833 MWt
800 Fuel Assemblies
248 Pins/Assembly
3.1E6 Fissions/SEC/WATT
1.19E20 Fissions/SEC at 100% Power
Reactor Coolant Volume (Cond 2) = 3E8ml
Suppression Pool Volume (Norm) = 4.5E9 ml
Containment NET FREE AIR VOL = 1.4E6 ft^3
Drywell NET FREE AIR VOL = 2.7E5 ft^3
Zironcium 178,400 lbm Total

6.0 DETAILS

NOTE

a. The results determined from this procedure are rough estimates of core damage and reactions based on these results should be conservative.

b. All analyses are based upon equilibrium full power core Iodine. If fuel damage is suspected to have occurred during times of reduced power or near the time of significant power change, the core Iodine inventory must be compensated accordingly by using Appendix I to calculate power correction factor Y.

c. Iodine concentrations may increase by a factor of 2 to 25 times above the equilibrium level due to spiking following a large power change or shutdown. This peak concentration normally occurs approximately 4 to 8 hours after the power change. Do not misinterpret this temporary change for fuel failure if there is no other evidence of fuel damage.

d. All values given are normalized to volumes of coolant at normal reactor coolant system pressure and temperature. To correct data, use Attachment II to determine factor X.

e. The determination of percent failed fuel is highly dependent on core temperature reached during the accident condition. Core temperatures in excess of 1600°F indicate possible cladding damage. Temperatures in excess of 4000°F indicate possible fuel melting.

f. If the isotopic analyses show the absence of Ruthenium and Tellurium, then assume that fuel melting has not occurred.

g. Due to analyses of past reactor accidents, all released Iodine is assumed to remain in the reactor coolant and/or the suppression pool; therefore, the total Iodine released is the summation of the two volumes and activities if the suppression pool was used during the accident.

h. Core damage below 1 percent is assumed to be a non-accident condition.

i. Addition 1 monitored parameters will be added to this procedure as information becomes available.

6.1 Post Accident sampling will be performed in accordance with 08-S-04-905, "Post Accident Sampling/Analyses". Direction as to parameters monitored are included in that instruction. The results obtained from the Containment H_2 and the coolant system I-131 analyses may be used in this procedure to estimate Core Damage.

6.2 Containment Hydrogen concentration obtained from plant process instrumentatin (J001A, J002A, J001B, J002B) or the Post Accident Sample Panel Monitor may be used to complete Attachment III to determine the extent of fuel cladding damage. If the Hydrogen igniters have been energized, the containment oxygen concentratin must be obtained from the Post Accident Sample Panel Oxygen Monitor (to be installed) and entered in the calculations. Hydrogen production due to Radiolysis is assumed to be insiginficant.

6.3 Iodine-131 test results performed by on-line instrumentation or laboratory analysis will be entered on Data Sheet IV. If the suppression pool was used as an extension of the Reactor System, then test results of both systems should be included. Convert the measured activity to a total I-131 concentration released. Use correction factors obtained for X and Y from Attachments I and II to normalize the data for comparison.

6.3.1 The data is then compared to the 100 percent I-131 release concentration obtained as follows:

a. 1.19E20 Fissions at 100% Power (3833 Mwt)
2.8% I-131 Fission Yield at Equilibrium

b. A = (F/S) (Yield)
= (1.19E20) (.028)
= 3.3E18 dps/3.7E10dpc
= 8.9E7Ci i-131

IODINE-131 INVENTORY POWER CORRECTION (4)

Situation 1

To correct for core Iodine Inventory if fuel damage is suspected to have occurred during times of any power level except 0% where the power level has not changed greater than ± 10% within the last 22 days, use the following equation:

$$Y = \frac{100}{\text{\% Full Power at time of failure}}$$

where Y is the correction factor to be used.

Example: The plant has been at 35% full power for the last 30 days when fuel damage is suspected. Therefore:

$$Y = \frac{100}{35} = 2.86$$

Situation 2

To correct for core iodine if fuel damage is suspected to have occurred at times other than fit Situation 1 above, use the following equation:

$$Y = \frac{100}{\text{old power level in \% } (e^{-\lambda t}) + \text{new power level in \% } (1-e^{-\lambda t})}$$

where:

Y = correction factor to be used.

old power level in % = the % full power before the power change.

new power level in % = the % full power after the power change at which time the fuel failure has occurred.

I = is the decay constant for I_{131} which equals .0864 day $^{-I}$

t = is the median time to make a power change plus the time after the power change until damage is suspected to have occurred in days.

Example: If it took 2 hours to make a power change and damage was suspected 10 hours fter the power change:

$$t = \frac{2}{2} + 10 = 11 \text{ hours}$$

DENSITY CORRECTION FACTOR, X, FOR SAMPLE TEMPERATURE CHANGES

Determine the appropriate Reactor Coolant temperature at the time of sampling. Normal Reactor Coolant System sample temperature is approximately 90°F. The intersection of both numbers is the density correction factor, X.

Reactor Cooland Sample Temperature °F

		90	
100		.998	
150		.985	
200		.968	
250		.947	
300		.923	
350		.895	
400		.864	
450		.825	
500		.788	
550		.740	
560		.729	
570		.718	
580		.708	
590		.694	
600		.681	

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HYDROGEN DATA CALCULATION SHEET

Hydrogen -

Monitor Readings: _____ J001A

NOTE: Use the monitor with the highest H_2 reading.

_____ J002A

_____ J001B

_____ J002B

_____ PASM

Oxygen -

Monitor Reading _____ PASM

Percent Core Cladding Reacted -

a. From H_2 Data

(%H_2 Monitor Reading) (1.67E6) = SCF H_2

(_____) (1.67E6) = SCF H_2

_____ SCF H_2

b. From O_2 Data

NORMAL (%O_2 Monitor Reading) - POST ACCID (O_2 Monitor Reading) = 0 depletion

_____ - _____ = _____

(O_2 depletion) (2) (1.674E6) = SCF H_2

_____ = SCF H_2

c. TOTAL VOLUME of H_2 liberates

SCF H_2 + SCF H_2 (O_2) = TOTAL SCF H_2

_____ + _____ = TOTAL SCF H_2

_____ = TOTAL SCF H_2

d. TOTAL MASS OF Zirconium reacted

$$= \frac{\text{Total SCF } H_2}{8.0 \text{ SCF } H_2/\text{lbm Zr reacted}}$$

= _____ lbm

e. Percentage of Core Cladding Reacted

$$\% = \frac{\text{_____ lbm Zr Reacted}}{\text{1.784E5 lbm Zr in Core}} \times 100$$

% = _____

IODINE-131 DATA CALCULATION SHEET

a. Iodine-131 Measured Measured Suppression Pool

(MAY BE N/A if Suppression Pool NOT USED)
(_____ uc/ml) (4.5E9 ml) (1E-6) = Ci(SP)

b. Iodine-131 Measured Reactor
(_____ uc/ml) (3E8 ml) (1E-6 ci/uc) = _____ Ci(Rx)

c. TOTAL I-131 Released

_____ Ci suppression Pool + _____Ci(Rx) = TOTAL I-131

_____ = TOTAL I-131

d. Power & Density Correction

(TOTAL I-131 Released) (Y) (X) = _____Ci Released Adjusted for 100% Pwr

e. Percent failed fuel

_____Ci Released/8.9E7 Ci Available = _____%

[1] Volumes may require density correction depending on system temperatures at time of sampling.

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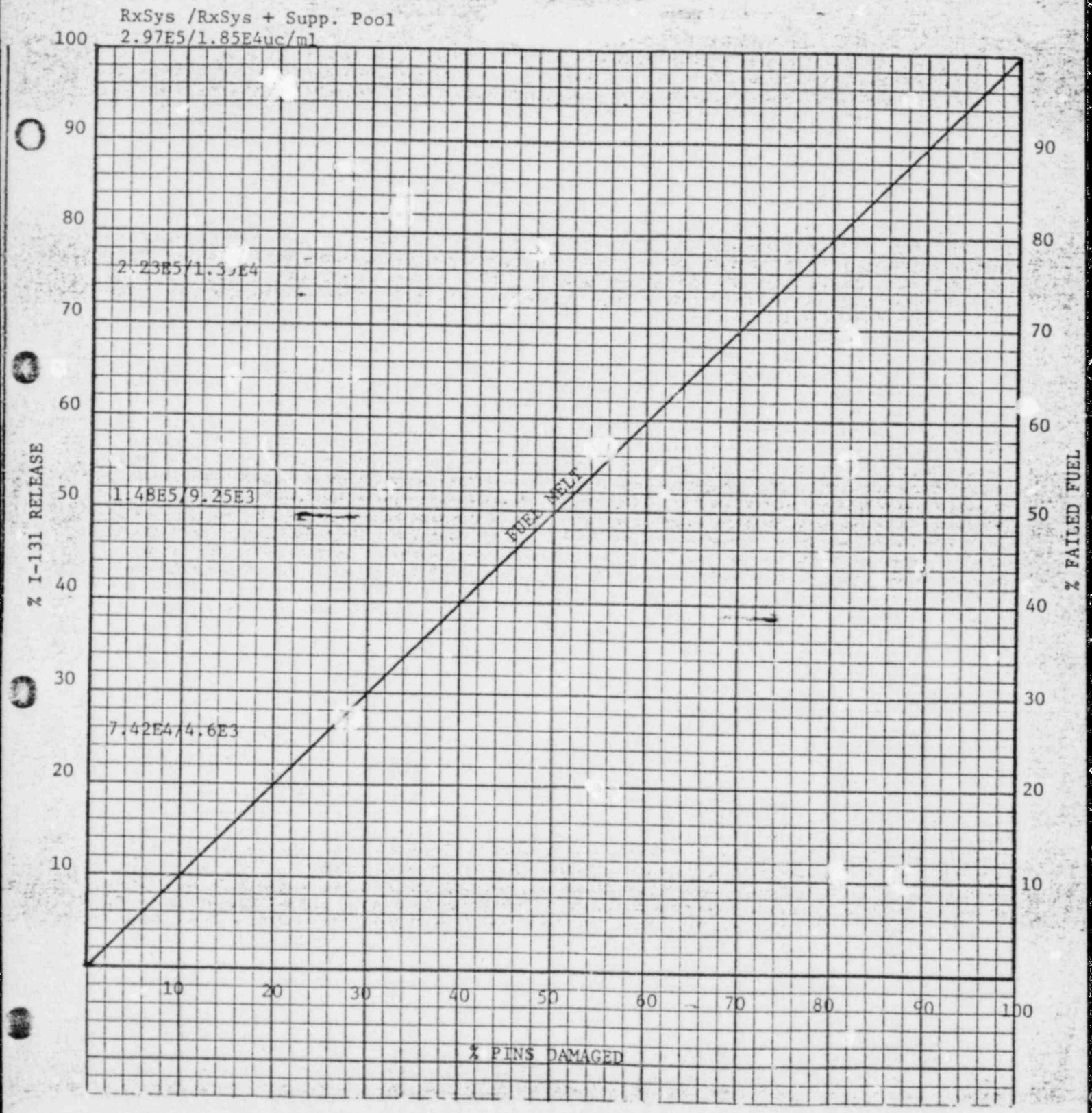
CORE DAMAGE GRAPH



BRANCH: Containment Systems Branch

CONCERN: For containment penetration 69, branch test line 3/4-HBB-188 located outboard of primary containment has only one isolation valve (P11F132). In order to comply with the applicable general design criteria for isolation we are requiring that an additional series valve be provided.

See also Grand Gulf SER outstanding issue, Item 1.9(8).

RESPONSE: An additional series valve will be added to the subject branch test line in accordance with the appropriate guidelines. FSAR Tables 6.2-44, 6.2-49, and Figure 6.2-78 will also be revised to reflect the design intent described above.

The above information will be incorporated into the next available FSAR amendment.

Branch: Instrumentation and Control Systems Branch

Concern: The staff's review indicated that containment spray system "B" could be manually initiated only by actuating the manual initiation switch continuously for 90 seconds. Although it may be desirable to maintain a time delay between initiation of the A and B loops, we consider such sustained operator action to be an unacceptable way to achieve this. Therefore, we require that the applicant initiate an engineering evaluation to determine the basis for the existing design. If the evaluation does not establish a justifiable design basis for the time delay, the applicant will propose a design modification prior to fuel load which will be implemented before startup after the first refueling outage.

See also Grand Gulf SER Confirmatory issue, Item 1.10(15).

Response: The requested engineering review of the existing design has been completed. The 90 second delay between train "A" and train "B" is required to prevent the rapid depressurization of containment that may result from the simultaneous initiation of both spray trains. Mississippi Power & Light does agree that the sustained operator action should not be required to manually initiate train "B". Therefore, the current design will be modified as necessary to delete the 90 second sustained hold down requirement. This modification, however, will retain the 90 second delay between train "A" and train "B" initiation. The proposed design details will be furnished for staff review when available and implemented prior to startup from the first regular refueling outage.