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

June 12, 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 Nos. 50-416 and 50-417
File 0260/0277/0755/L-860.0
Response to TMI Requirements
AECM-81/153

Attached is Mississippi Power & Light Company's response to the NUREG-0737 requirements that are applicable to Grand Gulf Nuclear Station.

The attached material will appear as Chapter 18 of a forthcoming FSAR amendment. The appropriate FSAR pages that require changes due to the addition of Chapter 18 are also included so that a complete review against approved requirements is possible. It should be noted that Chapter 18 references Section 13.1 frequently. Section 13.1 has recently been provided to the NRC by letter AECM-81/193, dated June 4, 1981.

Your expeditious review of this submittal will allow us to resolve any questions and provide for an uninterrupted licensing process.

If there are any questions concerning this response, please contact us.

Yours truly,

John D Richardson for
L. F. Dale
Manager of Nuclear Services

WWK/SHH/JDR:lm
Attachment

cc: (See Next Page)

B001
S 1/1

A

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

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

18. RESPONSES TO TMI RELATED REQUIREMENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

18.0 Introduction

This chapter contains a response for each TMI-related requirement identified in NUREG-0737 and applicable to Grand Gulf Nuclear Station.

18.1 Response to Requirements of NUREG-0737

18.1.1 Shift Technical Advisor (I.A.1.1)

REQUIREMENT

Each licensee shall provide an on-shift technical advisor to the shift supervisor. The Shift Technical Advisor (STA) may serve more than one unit at a multiunit site if qualified to perform the advisor function for the various units.

The STA shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The STA shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the STAs that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

RESPONSE

The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have a minimum of 12 months of nuclear power plant experience. Further discussion of Shift Technical Advisor qualifications is found in Section 13.1.3.1.7; the STA training program is discussed in Section 13.2.1.2.10; and, the duties and responsibilities of the STA are discussed in Section 13.1.2.2.3.4.1.

18.1.2 Shift Supervisor Administrative Duties (I.A.1.2)

REQUIREMENT

Review the administrative duties of the shift supervisor and delegate functions that detract from or are subordinate to the management responsibility for assuring safe operation of the plant to other personnel not on duty in the control room.

RESPONSE

Each operating shift has both a Shift Superintendent and a Shift Supervisor assigned. The Shift Superintendent is the Plant Manager's direct management representative and as such is responsible for the bulk of the administrative duties outside the control room. The Shift Supervisor reports to the Shift Superintendent and is responsible for

the actual operation of his assigned unit during his shift. Further discussion of the duties and responsibilities of the Shift Supervisor is contained in the response to Item I.C.3 and in Section 13.1.2.2.3.2.

The administrative duties of the Shift Supervisor will be periodically reviewed by the Plant Manager. Administrative duties that detract from the Shift Supervisor's responsibility for assuring the safe operation of the unit will be delegated to others not assigned to control room duties.

18.1.3 Shift Manning (I.A.1.3)

REQUIREMENT

Licensees of operating plants and applicants for operating licenses shall include in their administrative procedures (required by license conditions) provisions governing required shift staffing and movement of key individuals about the plant. These provisions are required to assure that qualified plant personnel to man the operational shifts are readily available in the event of an abnormal or emergency situation.

The administrative procedures shall also set forth a policy, the objective of which is to operate the plant with the required staff and develop working schedules such that use of overtime is avoided, to the extent practicable, for the plant staff who perform safety-related functions (e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, I&C technicians and key maintenance personnel).

RESPONSE

Grand Gulf Nuclear Station Administrative Procedures address shift manning and limitations on overtime as follows:

Minimum On-Duty Operations Shift Composition[1]
(Unit 1 Prior to Receipt of Operating License for Unit 2)

Single Unit Operating

Senior Reactor Operator	2
Reactor Operator	2
Non-Licensed Operator	2

Single Unit Shutdown

Senior Reactor Operator	1
Reactor Operator	1
Non-Licensed Operator	1

(Units 1 & 2 After Receipt of Operating License for Unit 2)

Both Units Operating

Senior Reactor Operator	3
Reactor Operator	3
Non-Licensed Operator	3

One Unit Operating, One Unit Shutdown

Senior Reactor Operator	2
Reactor Operator	3
Non-Licensed Operator	3

Both Units Shutdown

Senior Reactor Operator	1
Reactor Operator	2
Non-Licensed Operator	3

NOTE

(1) The numbers do not include the Senior Reactor Operator, or Senior Reactor Operator Limited to fuel handling, in charge of supervising core alterations on a particular unit.

Operating: Includes operational conditions defined in GGNS Technical Specifications as POWER OPERATION, STARTUP, and HOT SHUTDOWN.

Shutdown: Includes operational conditions defined in GGNS Technical Specifications as COLD SHUTDOWN and REFUELING.

Maximum Working Hours

a. An individual should not work more than 12 consecutive hours.

b. If a reactor operator is required to work in excess of 8 continuous hours, he shall be periodically relieved of primary duties at the control board such that periods of duty at the control board do not exceed approximately 4 hours.

c. There should be at least a 12-hour break between all work periods.

d. An individual should not work more than 72 hours in any 7-day period.

e. An individual should not work more than 14 consecutive days without having 2 consecutive days off.

f. In the event that special circumstances arise that require deviation from the above, such deviations should be authorized by the Plant Manger with appropriate documentation of the cause.

g. If an operator is required to work in excess of 12 continuous hours, his duties should be carefully selected. It is preferable that he not be assigned any task that affects core reactivity or could possibly endanger the safe operation of the plant.

h. If a Reactor Operator or Senior Reactor Operator has been working more than 12 hours during periods of extended shutdown (e.g., at duties away from the control board), such individuals shall not be assigned shift duty in the control room without at least a 12-hour break preceding such an assignment.

Additional information concerning shift manning is discussed in Section 13.1.2.1.

18.1.4 Immediate Upgrading of Reactor Operator and Senior Reactor Operator Training and Qualifications (I.A.2.1)

REQUIREMENT

Applicants for Senior Reactor Operator (SRO) license shall have 4 years of responsible power plant experience, of which at least 2 years shall be nuclear power plant experience (including 6 months at specific plant) and no more than 2 years shall be academic or related technical training. After fuel loading applicants shall have 1 year of experience as a licensed operator or equivalent.

Certifications that operator license applicants have learned to operate the controls shall be signed by the highest level of corporate management for plant operation.

Applicants must revise training programs to include training in heat transfer, fluid flow, thermodynamics, and plant transients.

RESPONSE

Applicants for a Senior Reactor Operator license at the Grand Gulf Nuclear Station have at least four years power plant experience. Prior to initial plant criticality at least one year of this power plant experience shall be nuclear power plant experience. After initial plant criticiality each senior licensed operator candidate shall have at least two years of nuclear power plant experience. A maximum of two years of power plant experience may be fulfilled by academic or related technical training on a one-for-one basis.

Prior to initial plant cirticality, the applicant shall possess six months of nuclear power plant experience at the Grand Gulf Nuclear Station. This experience may include participation in system acceptance, preoperational testing, and/or writing/verification of plant operating procedures.

After initial plant criticality, the applicant shall possess at least six months of nuclear power plant experience at the Grand Gulf Nuclear Station. The applicant's onsite experience shall include either:

a. If the applicant does not hold or has not held an NRC Operator's License on GGNS, the individual shall spend three months on shift carrying out the duties of the Control Room Operator while under the direct supervision of the Licensed Control Room Operator. The applicant shall also spend an additional three months on shift carrying out the duties of the shift supervisor while under the direct supervision of the licensed shift supervisor.

b. If the applicant holds or has held an NRC Operator License on GGNS, the individual shall spend three months on shift carrying out the duties of the shift supervisor while under the direct supervision of the licensed shift supervisor.

Additionally, after initial plant criticality, senior licensed operator applicants shall have held an NRC Operator License for a period of one year.

The highest level of corporate management for plant operations is the Assistant Vice President - Nuclear Production; and, is responsible for certification of the competency of each license applicant to operate the plant safely and competently prior to proposing the candidate for licensing by the NRC. This certification shall include considerations of successful completion of training, demonstrated abilities, satisfactory health, dependability, and stability.

Licensed Operator Training and Qualification programs at GGNS include training in heat transfer, fluid flow, thermodynamics, and plant transients.

18.1.5 Administration of Training Programs (I.A.2.3)

<u>REQUIREMENT</u>

Pending accreditation of training institutions, training instructors who teach systems, integrated response, transient and simulator courses shall successfully complete a Senior Reactor Operator (SRO) examination prior to fuel loading and instructors shall attend appropriate retraining programs that address, as a minimum, current operating history, problems and changes to procedures and administrative limitations. In the event an instructor is a licensed SRO, his retraining shall be the SRO requalification program.

<u>RESPONSE</u>

Instructors for the various onsite training lectures will be supplied by the Grand Gulf Training Section or consultants. Selection of the particular individual to conduct a specific training lecture will be based upon individual availability and knowledge of the subject matter involved. Permanent Training Center instructors and consultants assigned to training, who, after initial criticality will teach systems, integrated response, transients and simulator courses to license candidates or NRC licensed personnel, shall either demonstrate or have previously demonstrated their competence to the NRC by successful

to H. Denton's letter of March 28, 1980. The program includes normal, abnormal, and emergency instruction on a simulator; and is further discussed in Section 13.2.2.2.

18.1.7 Independent Safety Engineering Group (I.B.1.2)

REQUIREMENT

Each applicant for an operating license shall establish an onsite independent safety engineering group (ISEG) to perform independent reviews of plant operations.

The principal function of the ISEG is to examine plant operating characteristics, NRC issuances, Licensing Information Service advisories, and other appropriate sources of plant design and operating experience information that may indicate areas for improving plant safety. The ISEG is to perform independent review and audits of plant activities including maintenance, modifications, operational problems, and operational analysis, and aid in the establishment of programmatic requirements for plant activities. Where useful improvements can be achieved, it is expected that this group will develop and present detailed recommendations to corporate management for such things as revised procedures or equipment modifications.

Another function of the ISEG is to maintain surveillance of plant operations and maintenance activities to provide independent verification that these activities are performed correctly and that human errors are reduced as far as practicable. ISEG will then be in a position to advise utility management on the overall quality and safety of operations. ISEG need not perform detailed audits of plant operations and shall not be responsible for sign-off functions such that it becomes involved in the operating organization.

RESPONSE

The Nuclear Plant Engineering organizaton of the Grand Gulf Nuclear Station functions primarily as an onsite independent safety engineering group (ISEG) to perform independent review of plant operations, and is discribed in further detail in Section 13.1.1.1.3.1.

18.1.8 Guidance for the Evaluation and Development of Procedures for Transients and Accidents (I.C.1)

REQUIREMENT

Reanalysis of small break LOCAs, transients, accidents, and inadequate core cooling and preparation of guidelines for development of emergency procedures should be completed and submitted to the NRC for review. The NRC staff will review the analyses and guidelines and determine their acceptability, and will issue guidance to licensees on preparing emergency procedures from the guidelines.

RESPONSE

Mississippi Power & Light Company has participated in the BWR Owners' Group program to develop emergency procedure guidelines for General Electric boiling water reactors.

In a letter dated June 30, 1980, Mr. R. H. Buchholz forwarded the GE Emergency Procedure Guidelines for the BWR 1-5 product lines to Mr. D. G. Eisenhut. Mr. Eisenhut in a letter dated October 21, 1980 informed the BWR Owners' Group that the guidelines were acceptable for trial implementation on six NTOL plants. These plants were either BWR-4 or BWR-5 product lines. MP&L participated with the Owners' Group in extending the guidelines to address BWR-6/Mark III plants, and on January 31, 1981 in a letter from Mr. D. B. Walters to Mr. D. G. Eisenhut, these revised guidelines were transmitted to the NRC. On January 27, 1981, MP&L provided to the NRC by letter (AECM-81/44) the Grand Gulf Nuclear Station (GGNS) emergency procedures which were written based on the revised BWR Emergency Procedures Guidelines.

In the Clarification of the NUREG-0737 requirement "for reanalysis of transients and accidents and inadequate core cooling and preparation of guidelines for development of emergency procedures," NUREG-0737 states:

> "Owners' group or vendor submittals may be referenced as appropriate to support this reanalysis. If owners' group or vendor submittals have already been forwarded to the staff for review, a brief description of the submittals and justification of their adequacy to support guideline development is all that is required."

Following are brief descriptions of the submittals to date and a justification of their adequacy to support guideline development:

A. Description of Submittals

(1) NEDO-24708A, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," December, 1980.

(a) Section 3.1.1 (Small Break LOCA).

Description and analysis of small break loss-of-coolant events, considering a range of break sizes, locations, and conditions, including equipment failures and operator errors; description and justification of analysis methods.

(b) Section 3.2.1 (Loss of Feedwater)

Description and analysis of loss of feedwater events, including cases involving stuck-open relief valves, and including equipment failures and oper[illegible] errors; description and justification of analy[illegible] methods.

(c) Section 3.2.2 (Other Operational Transients)

Description and analysis of each FSAR Chapter 15 event resulting in a reactor system transient; demonstration of applicability of analyses of Sections 3.1.1, 3.2.1, and 3.5.2.1 to each event; demonstration of applicability of Emergency Procedure Guidelines to each event.

(d) Section 3.3 (BWR Natural and Forced Circulation)

Description of natural and forced circulation cooling; factors influencing natural circulation, including noncondensibles; reestablishment of forced circulation under transient and accident conditions.

(e) Section 3.5.2.1 (Analyses to Demonstrate Adequate Core Cooling)

Description and analysis of loss-of-coolant events, loss of feedwater events, and stuck-open relief valve events, including severe multiple equipment failures and operator errors which, if not mitigated, could result in conditions of inadequate core cooling.

(f) Section 3.5.2.3 (Diverse Methods of Detecting Adequate Core Cooling)

Description of indications available to the BWR operator for the detection of adequate core cooling (detailed instrument response are described in Sections 3.1.1, 3.2.1, and 3.5.2.1).

(g) Section 3.5.2.4 (Justification of Analysis Methods)

Description and justification of analysis methods for extremely degraded cases are treated in Section 3.5.2.1.

(2) BWR Emergency Procedure Guidelines (Revision 1) - submitted in prepublication form January 31, 1981.

Guidelines for BWR Emergency Procedures based on identification and response to plant symptoms; including a range of equipment failures and operator errors; including severe multiple equipment failures and operator errors which, if not mitigated, would result in conditions of inadequate core cooling; including conditions when core cooling status is uncertain or unknown.

B. Adequacy of Submittals

The submittals described in paragraph A have been discussed and reviewed extensively among the BWR Owners' Group, the General Electric Company, and the NRC staff. The NRC staff has found (NUREG-0737 p. I.C.1-3) that "the analysis and guidelines submitted by the General Electric Company (GE) Owners' Group...comply with the requirements [of the NUREG-0737 clarification]." In Reference 1, the Director of the Division of Licensing states, "we find the Emergency Procedure Guidelines acceptable for trial implementation [on six plants with applications for operating licenses pending]."

MP&L believes that in view of these findings, no further detailed justification of the analyses or guidelines is necessary at this time.

Reference 1 further states, "during the course of implementation we may identify areas that require modification or further analysis and justification." The enclosure to Reference 1 identifies several such areas. MP&L will work with the BWR Owners' Group in responding to such requests.

By our commitment to work with the Owners' Group on such requests, on schedules mutually agreed to by the NRC and the Owners' Group, and by reference to the BWR Owners' Group analyses and guidelines already submitted, our response to the NUREG-0737 requirement "for reanalysis of transients and accidents and inadequate core cooling and preparation of guidelines for development of emergency procedures" is complete.

C. References

(1) Letter, D. G. Eisenhut (NRC) to S. T. Rogers (BWR Owners' Group), regarding Emergency Procedure Guidelines, October 21, 1980.

(2) Letter, R. H. Buchholz (GE) to D. G. Eisenhut (NRC), regarding Emergency Procedure Guidelines, June 30, 1980.

18.1.9 Shift Relief and Turnover Procedures (I.C.2)

REQUIREMENT

Revise plant procedures for relief and turnover to require signed checklists and logs to assure that the operating staff (including auxiliary operators and maintenance personnel) possess adequate knowledge of critical plant parameter status, system status, system availability, system alignment and systems (or components) that are in a degraded mode of operation permitted by the Technical Specifications.

A system shall be established to evaluate the effectiveness of the shift and relief turnover procedures (for example, periodic independent verification of system alignments).

RESPONSE

GGNS Administrative Procedure 02-S-01-4, Shift Relief and Turnover, requires the on-coming Control Room Operator to perform a visual check of all critical plant parameters, the availability and proper alignment of all the emergency core cooling systems in the control room, and a general status of the control boards. He must fill out and submit a Status Checksheet to the on-coming Shift Supervisor and the Shift Superintendent for their review and signature at the beginning of his shift.

The Status Checklist lists the critical plant parameters, the systems which are to be checked at the control consoles, allowed inoperable times as stated in the Technical Specifications, and cumulative inoperable times.

Additionally, Auxiliary Operators will use an Area Turnover Sheet during their shift turnover. The off-going operator will use the turnover sheet to inform the on-coming operator of the area status including system/components degraded or inoperable, evolutions in progress, and any abnormal conditions.

The effectiveness of these shift relief and turnover procedures will be measured by Corporate Quality Assurance audits in accordance with the NRC accepted Grand Gulf Nuclear Station Operational Quality Assurance Manual (Topical MPL-TOP-1).

18.1.10 Shift Supervisor Responsibilities (I.C.3)

REQUIREMENT

Revise plant procedures to assure that duties, responsibilities, and authority of the shift supervisor and control room operators are properly defined.

RESPONSE

The duties, responsibilities, and authority of Grand Gulf Nuclear Station shift supervisors and control room operators is discussed in Sections 13.1.2.2.3.2 and 13.1.2.2.3.3. Further, GGNS Administrative Procedure 01-S-06-2, Conduct of Operations, defines the responsibility and authority of the Shift Supervisor and licensed operators.

18.1.11 Control Room Access (I.C.4)

REQUIREMENT

Revise plant procedures to limit access to the control room to those individuals responsible for the direct operation of the plant, technical advisors, specified NRC personnel, and to establish a clear line of authority, responsibility, and succession in the control room.

RESPONSE

Grand Gulf Nuclear Station Administrative Procedure 01-S-06-4, Access and Conduct in the Control Room, limits access to the control room to those individuals responsible for operation of the plant, technical assistance called in to support that operation, and others as deemed necessary. Additionally, the control room is to be kept clear except for on-duty Operations personnel and access is not allowed without the permission of the Shift Supervisor or the Reactor Operator at the controls.

GGNS Administrative Procedure 01-S-06-2, Conduct of Operations, delineates the line of authority, responsibility, and succession inside and outside the control room in the following manner:

a. The Plant Manager has the overall responsibility for operation of GGNS.

b. When the Plant Manager is not available to supervise the operation of GGNS, the Assistant Plant Manager shall assume this responsibility and authority for safe and efficient operation of the station. The Operations Superintendent shall assume this responsibility and authority if neither the Plant Manager nor the Assistant Plant Manager is available.

c. In the absence of the persons designated above, the Operations Shift Superintendent shall assume overall responsibility and authority for operation of the station.

d. The Shift Superintendent is, at all times, the Plant Manager's direct management representative for the conduct of operations and, as such, has the responsibility and authority to direct all activities and personnel at GGNS as required to:

   (1) Protect the health and safety of the public and the environment.

   (2) Protect the health and safety of GGNS employees, contractors, or other personnel onsite.

   (3) Prevent damage to GGNS equipment and structures.

   (4) Protect the physical security of GGNS.

   (5) Ensure compliance with the GGNS Operation License.

e. The Shift Superintendent shall retain this responsibility and authority, unless he is formally relieved of:

   (1) Operating responsibilities by a Licensed Senior Reactor Operator at the direction of any of the following personnel, or by any of the following personnel should they be Licensed Senior Reactor Operator:

      (a) Operations Superintendent

(b) Assistant Plant Manager

(c) Plant Manager

(2) Accident management responsibilities as described in the GGNS Emergency Plan.

f. The Shift Superintendent shall seek the advice and guidance of responsible GGNS personnel in executing his duties whenever in doubt as to the proper course of action and shall promptly inform responsible supervisors of significant actions affecting their responsibilities.

g. The Shift Supervisors are responsible to the Shift Superintendent for the conduct of operations of their units and, as such, have responsibility and authority to direct all activities and personnel in their unit during normal operations and emergencies as necessary to:

(1) Protect the health and safety of the public and the environment.

(2) Protect the health and safety of employees, contractors, or other personnel in their unit.

(3) Prevent damage to their unit's equipment and structures.

(4) Ensure compliance with their unit's operating license.

h. The Shift Supervisor has the responsibility to maintain an overall "big picture" concept of the unit operations and not to become totally envolved in any single plant operation during times of an emergency when multiple operations are required.

i. The Shift Supervisor shall retain this responsibility and authority under the direction of the Shift Superintendent unless he is formally relieved by a licensed Senior Reactor Operator at the direction of the following personnel or by the following personnel should they hold a valid Senior Reactor Operators license:

(1) Shift Superintendent

(2) Operations Superintendent

(3) Assistant Plant Manager

(4) Plant Manager

j. If the Shift Superintendent becomes incapacitated the Unit 1 Shift Supervisor will perform the functions of the Shift Superintendent until relieved.

18.1.12 Procedures for Feedback of Operating Experience to Plant Staff (I.C.5)

<u>REQUIREMENT</u>

Review administrative procedures to assure that operating experience from within and outside the organization is continually provided to operators and other operational personnel and is incorporated in training programs.

<u>RESPONSE</u>

Section 18.1.7, Independent Safety Engineering Group, addressed the function and responsibilities of the GGNS Nuclear Plant Engineering organization. An additional responsibility of this organization is to ensure operating experience information pertinent to plant operations is supplied to plant staff personnel and is incorporated into training and retraining programs in a timely manner.

GGNS Administrative Procedure 01-S-04-15, Required Reading Program, further details the persons responsible and the mechanism used to disseminate operating experience information to operations personnel.

18.1.13 Guidance on Procedures for Verifying Correct Performance of Operating Activities (I.C.6)

<u>REQUIREMENT</u>

It is required (from NUREG-0660) that licensees' procedures be reviewed and revised, as necessary, to assure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. This will reduce the frequency of occurrence of situations that could result in or contribute to accidents. Such a verification system may include automatic system status monitoring, human verification of operations, and maintenance activities independent of the people performing the activity (see NUREG-0585, Recommendation 5), or both.

<u>RESPONSE</u>

The Grand Gulf Nuclear Staion <u>Operational Quality Assurance Manual</u> (Topical MPL-TOP-1), which has been accepted for use by the NRC, endorses with some clarification Regulatory Guide 1.33 Revision 2, February 1978, which in turn endorses ANSI 18.7-1976.

The Grand Gulf Nuclear Station <u>Operations Manual</u> establishes the procedures necessary to implement he requirements of Regulatory Guide 1.33 and ANSI 18.7-1976. The following procedures have been written, approved, and implemented to verify correct performance of operating activities.

(a) 01-S-06-1, Protective Tagging System
(b) 01-S-06-2, Conduct of Operations
(c) 01-S-06-3, Control of Temporary System Alterations
(d) 01-S-06-12, GGNS Surveillance Program

GGNS procedures take the following actions to meet the requirements of the supplemental provisions set out in I.C.6:

1. The GGNS Operations Manual procedures cited above require the permission of the Shift Supervisor, who will be a licensed SRO, to release plant systems for maintenance and surveillance tests.

2. The Shift Supervisor alone has the authority to release systems and equipment for maintenance or surveillance testing or to return that equipment to service. GGNS procedures do not allow delegation of this function.

3. The Protective Tagging System procedure and the Control of Temporary System Alterations procedure both require independent verification of tags or temporary alterations on safety-related equipment or systems.

4. The Protective Tagging System procedure designates the control room operator as controller for all equipment under the tagging authority of the Shift Supervisor. As controller, the control room operator prepares the release requests, maintains the appropriate logs, and directs placement and removal of tags. He has similar responsibilities with respect to temporary alterations. These procedural controls ensure the operator is always aware of the status of the plant.

5. Independent verification is required to ensure safety-related equipment is properly returned to service if functional testing cannot be performed. In addition, the Protective Tagging System procedure requires the Shift Supervisor to designate functional check outs that may be required.

18.1.14 Nuclear Steam Supply System Vendor Review of Procedures (I.C.7)

REQUIREMENT

Obtain nuclear steam supply system vendor review of power-ascension and emergency operating procedures to further verify their adequacy.

RESPONSE

Mississippi Power & Light Company will have General Electric Company, the NSSS vendor, review low power test procedures prior to fuel loading and power ascension test procedures and emergency procedures prior to the issuance of a full power license. This review to verify procedure adequacy will be performed at Grand Gulf under the direction of the GE

Site Operations Manager. Test procedures will be reviewed by the GE Startup Test Engineers supervised by the GE Lead Engineer - Startup Test, Design, and Analysis (STD&A). Emergency procedures will be reviewed by the GE Operations Superintendents supervised by the GE Operations Superintendent - Startup Test Operation (STO).

18.1.15 Pilot Monitoring of Selected Emergency Procedures for NTOL Applicants (I.C.8)

REQUIREMENT

Correct emergency procedures, as necessary, based on the NRC audit of selected plant emergency operating procedures (e.g., small-break loss-of-coolant accident, loss of feedwater, restart of engineered safety features following a loss of AC power and steam-line break).

RESPONSE

As a result of an NRC audit, if so required, Mississippi Power & Light Company will revise the Grand Gulf Nuclear Station emergency operating procedures.

18.1.16 Control Room Design Review (I.D.1)

REQUIREMENT

Perform a preliminary assessment of the control room to identify significant human factors and instrumentation problems and establish a schedule approved by the NRC for correcting deficiencies.

RESPONSE

Mississippi Power & Light Company contracted the Essex Corporation to perform a human factors evaluation of the Grand Gulf control room. The results of that study and MP&L's plans for corrective action were submitted to the NRC in a letter from Mr. L. F. Dale to Mr. H. R. Denton dated December 29, 1980 (AECM-80/316).

18.1.17 Plant Safety Parameter Display Console (I.D.2)

REQUIREMENT

Install a safety parameter display system (SPDS) that will display to operating personnel a minimum set of parameters which define the safety status of the plant. This can be attained through continuous indication of direct and derived variables as necessary to assess plant safety status.

RESPONSE

Mississippi Power & Light Company submitted to the NRC in a letter from Mr. L. F. Dale to Mr. H. R. Denton dated February 12, 1981 (AECM-81/52) a description and design specification for the Grand Gulf Emergency Response Facility Information System (ERFIS). The Safety Parameter Display System (SPDS) is included in that description.

18.1.18 Training During Low-Power Testing (I.G.1)

REQUIREMENT

Define and ommit to a special low-power testing program, approved by the NRC, to be conducted at power levels no greater than 5 percent for the purposes of providing meaningful technical information beyond that obtained in the normal startup test program and to provide supplemental training

RESPONSE

Mississippi Power & Light Company received on January 26, 1981, a letter from the NRC's Mr. Robert L. Tedesco providing additional guidance with regard to TMI Task Action Plan I.G.1 as it applied to BWR OL applications. MP&L responded to that letter with our program for meeting item I.G.1 requirements in a letter dated April 7, 1981, from Mr. L. F. Dale to Mr. Robert L. Tedesco (AECM-81/84).

18.1.19 Reactor Coolant System Vents (II.B.1)

REQUIREMENT

Each applicant and licensee shall install reactor coolant system (RCS) and reactor vessel head high point vents remotely operated from the control room. Although the purpose of the system is to vent noncondensible gases from the RCS which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a loss-of-coolant accident (LOCA) or a challenge to containment integrity. Since these vents form a part of the reactor coolant pressure boundary, the design of the events shall conform to the requirements of Appendix A to 10 CFR Part 50, "General Design Criteria." The vent system shall be designed with sufficient redundancy that assures a low probability of inadvertent or irreversible actuation.

RESPONSE

The primary method of venting the Reactor Pressure Vessel at Grand Gulf is through twenty (20) safety/relief valves located on the main steam lines between the reactor vessel and the first main steam isolation valve within the drywell. These power operated relief valves satisfy the intent of the NUREG-0737 requirement. Further information regarding the design, qualification, and power source of these valves has been provided in Sections 5.1, 5.2, 7.3, and 8.3.

RESPONSE

A radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials has been accomplished at Grand Gulf Nuclear Station. The results of that review, together with a description of the review, is presented in Section 12.6.

A discussion of the radiation and shielding review as it relates to environmental qualification of equipment will be provided in a latter FSAR amendment once the review of NUREG-0588 is complete.

18.1.21 Postaccident Sampling Capability (II.B.3)

REQUIREMENT

A design and operational review of the reactor coolant and containment atmosphere sampling line systems shall be performed to determine the capability of personnel to promptly obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 and 18-3/4 rem to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to promptly quantify (in less than 2 hours) certain radionuclides that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and nonvolatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly (i.e., the boron sample analyses within an hour and the chloride sample analysis within a shift).

RESPONSE

The capability to obtain and perform radioisotopic and chemical analyses of the reactor coolant and the containment atmosphere samples is provided by the Process Sampling System via the Postaccident Sampling Station, which is described in Subsections 7.7.1.11.4.2 and 9.3.2.2.4.

18.1.22 Training for Mitigating Core Damage (II.B.4)

REQUIREMENT

Licensees are required to develop a training program to teach the use of installed equipment and systems to control or mitigate accidents in which the core is severely damaged. They must then implement the training program.

RESPONSE

Mississippi Power & Light will utilize a degraded core training course provided by an outside source; and, is not yet fully developed. This course will be given to all shift technical advisors and operations personnel from the plant manager to and including licensed operators prior to fuel load.

Managers and technicians in instrumentation and controls, health physics, and chemistry will be given training commensurate with their responsibilities during accidents which involve severe core damage.

18.1.23 Performance Testing of Boiling-Water Reactor and Pressurized-Water Reactor Relief and Safety Valves (II.D.1)

REQUIREMENT

Pressurized-water reactor and boiling-water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design-basis transients and accidents.

RESPONSE

Mississippi Power & Light is participating in the BWR Owner's Group program to test safety/relief valves and is providing two Dikkers safety/relief valves for use in the testing. A description of the BWROG testing program was sent to the NRC on September 17, 1980 by a letter from D. B. Waters to R. N. Vollmer.

18.1.24 Direct Indication of Relief and Safety Valve Position (II.D.3)

REQUIREMENT

Reactor coolant system relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve-position detection device or a reliable indication of flow in the discharge pipe.

RESPONSE

Mississippi Power & Light will provide a safety/relief valve position monitoring system consisting of pressure switches, sensor relays, annuciators, and indicating lights as necessary to monitor, to annuciate, and to indicate the open/closed condition of each safety/relief valve.

The downstream piping of each safety/relief valve will be filled with a single hydraulic sensing line connected to a pressure switch. The pressure switch output will in turn be connected to a circuit borad assembly containing relays and electronic logic.

An open safety/relief valve pressurizes the discharge line and the hydraulic sensing line to the pressure switch, actuating the pressure switch.

Pressure switch actuation causes annuciation and indication in the control room. After the pressure in the discharge piping decays, the pressure switch resets.

A more complete discussion of the safety/relief valve monitoring system will be provided in a future FSAR amendment once final design and installation is complete.

18.1.25 Dedicated Hydrogen Penetrations (II.E.4.1)

REQUIREMENT

Plants using external recombiners or purge systems for postaccident combustible gas control of the containment atmosphere should provide containment penetration systems for external recombiner or purge systems that are dedicated to that service only, that meet the redundancy and single-failure requirements of General Design Criteria 54 and 56 of Appendix A to 10 CFR 50, and that are sized to satisfy the flow requirements of the recombiner or purge system.

The procedures for the use of combustible gas control systems following an accident that results in a degraded core and release of radioactivity to the containment must be reviewed and revised, if necessary.

RESPONSE

Grand Gulf Nuclear Station has internal hydrogen recombiners which are located inside the containment in combination with a drywell purge system (see Section 6.2.5). A backup filtered containment purge through a dedicated seismic Category I penetration is also provided. As internal systems located inside the containment, the only containment piping penetrations associated with the recombiners are the drywell and containment hydrogen analyzer sample and sample return lines. Each of these 3/4" lines has two remote manual motor operated isolation valves. Since these are essential penetrations (see II.E.4.2), it is required that these valves remain open.

The present system is designed based on hydrogen generation rate calculations using Regulatory Guide 1.7 (Revision 1). MP&L has a program underway to improve the capability of the Mark III containment in dealing with significant amounts of hydrogen well in excess of those considered under 10 CFR 50.44. A description of the hydrogen control program was provided in a letter from Mr. J. P. McGaughy to Mr. Robert L. Tedesco dated December 9, 1980 (AECM-80/300). Later, a letter discussing the ultimate capacity analysis of the Grand Gulf Mark III containments was provided on January 21, 1981 to Mr. Tedesco (AECM-81/38).

Mississippi Power & Light Company is working with the BWR Owners' Group and General Electric Company to develop an emergency procedure guideline for combustible gas control. Emergency procedures and system operating procedures will be reviewed and revised considering these guidelines and the results of the GGNS hydrogen control study.

18.1.26 Containment Isolation Dependability (II.E.4.2)

REQUIREMENT

(1) Containment isolation system designs shall comply with the recommendations of Standard Review Plan Section 6.2.4 (i.e., that there be diversity in the parameters sensed for the initiation of containment isolation).

(2) All plant personnel shall give careful consideration to the definition of essential and nonessential systems, identify each system determined to be essential, identify each system determined to be nonessential, describe the basis for selection of each essential system, modify their containment isolation designs accordingly, and report the results of the reevaluation to the NRC.

(3) All nonessential systems shall be automatically isolated by the containment isolation signal.

(4) The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

(5) The containment setpoint pressure that initiates containment isolation for nonessential penetrations must be reduced to the minimum compatible with normal operating conditions.

(6) Containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CSB 6-4 or the Staff Interim Position of October 23, 1979 must be sealed closed as defined in SRP 6.2.4, item II.3.f during operational conditions 1, 2, 3, and 4. Furthermore, these valves must be verified to be closed at least every 31 days.

(7) Containment purge and vent isolation valves must close on a high radiation signal.

RESPONSE

(1) Grand Gulf Nuclear Station complies with this requirement as stated in Subsection 7.2.1.1.2.4.1.1.5.

(2) A reevaluation of all systems penetrating the primary containment has been accomplished. The results of the reevaluation are listed in Table 18.1-1. A new classification called beneficial has been added for nonessential systems that are not required for accident mitigation, but are desirable for plant operation. See also Table 6.2-44.

(3) All nonessential power operated isolation valves are automatically closed upon receipt of a containment isolation signal. There are some locked closed manual valves and blind flanges in nonessential systems.

(4) A letter from Mr. Robert L. Tedesco to Mr. J. P. McGaughy, dated December 12, 1980, required MP&L to prepare a response to IE Bulletin No. 80-06 dealing with ESF reset logic. As a result of receiving this letter which included additional guidance for evaluating reset logic, MP&L is reviewing the design of the safety systems for automatic containment isolation logic. Partial response has been provided in letters from Mr. L. F. Dale to the NRC's Mr. H. R. Denton, dated February 20, 1981 (AECM-81/78) and June 1, 1981 (AECM-81/154).

(5) The containment isolation analytical setpoint pressure for Mark I, II and III containments is approximately 2 psig (drywell pressure). In the GGNS Standard Technical Specifications the trip setpoint is a value less than the analytical value. Under normal operating conditions, fluctuations in the atmospheric barometric pressure as well as heat inputs from such sources as pumps are expected to result in drywell pressure increases of approximately 1 psig. Consequently, the technical specification trip setpoint at a value less than 2 psig provides a 1 psig margin above the expected normal operating pressure. A 1 psig margin to isolation has proved on earlier operating plants to be a suitable value to minimize the

possibility of spurious containment isolation. At the same time, such a low value (particularly in view of the small drywell volume of the Mark III containment) provides a very sensitive and positive means of detecting and protecting against breaks and leaks in the reactor coolant system. In view of the guidelines set forth in the clarification to position 5 which suggest a maximum of 1 psig differential between the maximum expected normal operating pressure and the instrument setpoint, no change of the setpoint is necessary for the Grand Gulf containment.

(6) See subsections 9.4.7 and 6.2.4.3.3; and, responses to questions 021.02 and 311.8. Grand Gulf Nuclear Station is also evaluating the NRC position attached to their letter of December 12, 1980 on IEB 80-06 as discussed in response to item (4) above.

(7) See response to item (4) above.

18.1.27 Additional Accident-Monitoring Instrumentation

18.1.27.1 Noble Gas Effluent Monitor (II.F.1.1)

REQUIREMENT

Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions as well as during normal operating conditions. Multiple monitors are considered necessary to cover the ranges of interest.

(1) Noble gas effluent monitors with an upper range capacity of 10^5 uCi/cc (Xe-133) are considered to be practical and should be installed in all operating plants.

(2) Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (as low as reasonably achievable (ALARA) concentrations to a maximum of 10^5 uCi/cc (Xe-133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors should overlap by a factor of ten.

RESPONSE

Grand Gulf Nuclear Station provides for continuous monitoring of high level, postaccident, releases of radioactive noble gases, both during and following an accident, via the Containment Ventilation Monitoring Systems described in Section 11.5.2.2.4.

18.1.27.2 Sampling and Analysis of Plant Effluents (II.F.1.2)

REQUIREMENT

Because iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent

monitoring of radioiodines for the accident condition shall be provided with sampling conducted by absorption on charcoal or other media, followed by onsite laboratory analysis.

RESPONSE

Grand Gulf Nuclear Station provides for continuous sampling of plant gaseous effluent for postaccident releases of radioactive iodines and particulates via the Containment Ventilation Monitoring Systems described in Section 11.5.2.2.4.

18.1.27.3 Containment High-Range Radiation Monitor (II.F.1.3)

REQUIREMENT

In containment radiation-level monitors with a maximum range of 10^8 rad/hr shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be developed and qualified to function in an accident environment.

RESPONSE

Grand Gulf Nuclear Station provides for in containment, high range, radiation monitoring in both the containment and drywell area via the In Containment Area Radiation Monitoring System described in Sections 7.5.1.2.3.6 and 12.3.4.3.

18.1.27.4 Containment Pressure Monitor (II.F.1.4)

REQUIREMENT

A continuous indication of containment pressure shall be provided in the control room of each operating reactor. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and -5 psig for all containments.

RESPONSE

Grand Gulf Nuclear Station provides for continuous measurement and indication of containment and drywell pressure by using two wide-range, and two narrow-range, containment pressure transmitters; and, two wide-range drywell pressure transmitters, that are continuously recorded and displayed in the control room. Further discussion is provided in Section 7.5.1.2.3.1.

18.1.27.5 Containment Water Level Monitor (II.F.1.5)

REQUIREMENT

A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for PWRs and cover the range from the bottom to the top of the containment sump. A wide range instrument shall also be provided for PWRs and shall cover the range from the bottom of the containment to the elevation equivalent to a 600,000 gallon capacity. For BWRs, a wide range instrument shall be provided and cover the range from the bottom to 5 feet above the normal water level of the suppression pool.

RESPONSE

Grand Gulf Nuclear Station continuously monitors suppression pool level with two wide-range and two narrow-range level signals that are recorded in the control room. Further discussion is provided in Section 7.5.1.2.3.3.

18.1.27.6 Containment Hydrogen Monitor (II.F.1.6)

REQUIREMENT

A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure.

RESPONSE

Grand Gulf Nuclear Station provides for continuous recording of hydrogen concentration in the containment and drywell atmospheres in the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure. Further discussion is provided in Section 7.5.1.2.8.3.

18.1.28 Instrumentation for Detection of Inadequate Core Cooling (II.F.2)

REQUIREMENT

Licensees shall provide a description of any additional instrumentation or controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

RESPONSE

Mississippi Power & Light supports the BWR Owners' Group position that no additional instrumentation is needed to monitor inadequate cooling at the Grand Gulf Nuclear Station.

18.1.29 Office of Inspection and Enforcement Bulletins

18.1.29.1 Safety-Related Valve Position (II.K.1.5)

REQUIREMENT

Review all valve positions, positioning requirements, positive controls and related test and maintenance procedures to assure proper ESF functioning.

RESPONSE

A response to the above requirement was forwarded to the NRC in letter AECM-80/26, dated March 19, 1980, which responded to IE Bulletin 79-08. Below is the response to this particular item provided at that time.

A review of the emergency core cooling systems (ECCS) indicated that the system valves' positions are suitably controlled by the following means:

1. Automatic actuation of power operated valves within the system is provided to isolate the boundary/bypass paths and to align the system for proper operation. Main control room valve position indication is provided for these valves. The handswitches in the control room for these valves are spring return to the auto position to allow the valve to operate automatically if required.

2. Manual valves within the main flow path are provided with locking provisions to ensure correct valve positions. Manual valves which are not accessible during power operation (i.e., located in drywell) are also provided with main control room position indicating lights.

3. Manual valves on branch piping to the main flow piping are provided with locking provisions if incorrect valve position could affect system safety function. Exceptions are the piping high point vents, low point drains, and test connection valves which are verified procedurally to be aligned properly for operation.

   For the condensate storage tank piping to the suction of the HPCS and RCIC pumps, the manual isolation valve adjacent to the storage tank will be verified procedurally that it has been aligned properly.

For all other safety-related systems other than ECCS, the power operated valves have been equipped with handswitches having the spring return feature and have been equipped with position indication in the control

room. The manual valves in these systems will be verified procedurally that they have been aligned properly. These manual valves are not equipped with position indication in the control room.

All system P&IDs have been reviewed to verify that the valves are positioned correctly for proper operation of the safety-related system.

A revision to the Protective Tagging Procedure is being prepared to use miniature tags on control panels to avoid obscuring any active indicators on control panels.

The position of each manually operated valve will be identified in a valve line-up sheet. Valve line-up checks will be conducted as required by technical specifications to verify system flow paths. In addition, valve line-up checks will be conducted after each refueling outage and following any major work on a system. For safety related systems/components, this valve line-up will have independent verifications. Where appropriate, valves will be locked in the designated position to prevent inadvertent repositioning.

If valve positions are to be changed for surveillance purposes, the surveillance procedure will have steps requiring return to normal valve line-up prior to completion. Start and completion of surveillance procedures will be logged in the control room logbook.

When maintenance is performed on a safety related system which requires valves to be repositioned, administrative procedures governing conduct of maintenance will require:

1. The approval of the Shift Supervisor prior to performing maintenance to allow the Shift Supervisor to verify redundant flow paths, etc. prior to authorizing maintenance.

2. The maintenance work documents to specify post-maintenance functional checks or operability tests to verify system return to normal following maintenance activities.

When possible, Operations will perform a functional test or Surveillance Operability Test following maintenance on any safety related system. When such tests are not possible, a complete valve and electrical lineup will be performed within the tagged boundary and a partial functional test will be performed where possible to provide assurance that systems are in fact functional after maintenance.

System line-up changes, other than those covered by step-by-step procedures will be logged and abnormal line-ups will be covered during shift turnover.

During periodic tours, Operators and Supervisory personnel will conduct spot checks of fluid system and electrical line-ups.

18.1.29.2 Safety-Related System Operability Status Assurance (II.K.1.10)

REQUIREMENT

Review and modify, as required, procedures for removing safety-related sytems from service (and restoring to service) to assure operability status is known.

RESPONSE

A response to the above requirement was forwarded to the NRC in letter AECM-80/26, dated March 19, 1980, which responded to IE Bulletin 79-08. Below is the response to this particular item provided at that time.

An Operations Section procedure which provides guidelines for release of permanent plant equipment will specify functional testing of redundant safety-related systems prior to the intentional removal of any safety related system from service. This procedure will also specify functional testing of safety related systems upon return to service following maintenance. Release of all permanent plant equipment from service will be authorized by the Shift Supervisor. This procedure will specify that the Shift Supervisor will notify other operations personnel of the status of plant systems and that control room operators are aware of all safety related systems which are removed from service.

18.1.29.3 Proper Functioning of Heat Removal Systems (II.K.1.22)

REQUIREMENT

Describe the automatic and manual actions necessary for proper functioning of the auxiliary heat removal systems that are used when the main feedwater system is not operable.

RESPONSE

A response to the above requirement was forwarded to the NRC in letter AECM-80/26, dated March 19, 1980, which responded to IE Bulletin 79-08. Below is the response to this particular item provided at that time.

This response describes both automatic and manual actions necessary for proper functioning of the auxiliary heat removal systems. These systems are used when the main feedwater system is not operable. The procedures are described in summary form assuming the reactor is scrammed and isolated from the main condenser.

Automatic action provides abundant make-up water to the core for initial cooling. Long term core and containment cooling can be provided with few manual actions. Information is available to the operator in the control room to assist him in taking the required manual actions. Information in the control room permits the operator to verify that the objective of these actions is being achieved.

Drywell High Pressure
Containment High Radioactivity Levels
Suppression Pool High Temperature
Safety Relief Valve (SRV) Discharge High Temperature
High/Low Feedwater Flow Rates
High/Low Main Steam Flow
High Containment, Steam Tunnel, and Equipment Area
 Differential Temperatures
High Differential Flow-Reactor Water Cleanup System
Abnormal Reactor Pressure
High Suppression Pool Water Level
High Drywell and Containment Sump Fill and Pumpout Rate
Valve Stem Leakoff High Temperatures
Low RCIC Steam Supply Pressure
High RCIC Steam Supply Flow
Low Main Steam Line Pressure

An example of the use of this additional information by the operator is as follows: Drywell high pressure is an indirect indication of coolant loss. Coincident high suppression pool temperature further verifies a loss of reactor coolant. High SRV discharge temperature would pinpoint loss of coolant via an open valve.

Other instrumentation that can signal abnormal plant status but not necessarily indicative of loss of coolant are:

High Neutron Flux
High Process Monitor Radiation Levels
Main Turbine Status Instrumentation
Abnormal Reactor Recirculation Flow
High Electrical Current (Amperes) to Recirc Pump Motors

Operators will be instructed in use of other available information to initiate safety systems as a continuing part of training.

18.1.30 Final Recommendations of Bulletins and Orders Task Force

18.1.30.1 Report Safety and Relief Valve Failures Promptly and Challenges Annually (II.K.3.3)

REQUIREMENT

Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs or safety valves should be documented in the annual report.

RESPONSE

Mississippi Power & Light commits to prompt reporting of safety/relief valve failures and will report challenges to the safety/relief valves annually.

18.1.30.2 Separation of High-Pressure Coolant Injection and Reactor Core Isolation Cooling System Initiation Levels - Analysis and Implementation (II.K.3.13)

REQUIREMENT

Currently, the reactor core isolation cooling (RCIC) system and the high-pressure coolant injection (HPCI) system both initiate on the same low-water-level signal and both isolate on the same high-water-level signal. The HPCI system will restart on low water level but the RCIC system will not. The RCIC system is a low-flow system when compared to the HPCI system. The initiation levels of the HPCI and RCIC system should be separated so that the RCIC system initiates at a higher water level than the HPCI system. Further, the initiation logic of the RCIC system should be modified so that the RCIC system will restart on low water level. These changes have the potential to reduce the number of challenges to the HPCI system and could result in less stress on the vessel from cold water injecton. Analyses should be performed to evaluate these changes. The analyses should be submitted to the NRC staff and changes should be implemented if justified by the analyses.

RESPONSE

Mississippi Power & Light has endorsed the position of the BWR Owners' Group delineated in the letter from Mr. R. H. Buchholz to Mr. D. G. Eisenhut dated October 1, 1980. That position is basically that "...the current design is satisfactory, and a significant reduction in thermal cycles is not necessary"; and, "...no significant reduction in thermal cycles is achievable by separating the setpoints."

Modification of the initiation logic for automatic restart of the RCIC system on low water level is being incorporated into the Grand Gulf design and will be incorporated in a later FSAR amendment.

18.1.30.3 Modify Break-Detection Logic to Prevent Spurious Isolation of High-Pressure Coolant Injection and Reactor Core Isolation Cooling (II.K.3.15)

REQUIREMENT

The high-pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) systems use differential pressure sensors on elbow taps in the steam lines to their turbine drives to detect and isolate pipe breaks in the systems. The pipe-break-detection circuitry has resulted in spurious isolation of the HPCI and RCIC systems due to the pressure spike which accompanies startup of the systems. The pipe-break-detection circuitry should be modified to that pressure spikes resulting from HPCI and RCIC system initiation will not cause inadvertent system isolation.

RESPONSE

The BWR Owners' Group has evaluated this issue and has recommended the addition of a time delay to the HPCI/RCIC break detection circuitry. Mississippi Power & Light has contracted with General Electric to provide this change to the Grand Gulf Nuclear Station RCIC steam line break detection circuity. A description of this change will be included in a latter FSAR amendment.

18.1.30.4 Reduction of Challenges and Failures of Relief Valves - Feasibility Study and System Modification (II.K.3.16)

REQUIREMENT

The record of relief-valve failures to close for all boiling-water reactors (BWRs) in the past 3 years of plant operation is approximately 30 in 73 reactor-years (0.41 failures per reactor-year). This has demonstrated that the failure of a relief valve to close would be the most likely cause of a small-break loss-of-coolant accident (LOCA). The high failure rate is the result of a high relief-valve challenge rate and a relatively high failure rate per challenge (0.16 failures per challenge). Typically, five valves are challenged in each event. This results in an equivalent failure rate per challenge of 0.03. The challenge and failure rates can be reduced in the following ways:

(1) Additional anticipatroy scram on loss of feedwater,

(2) Revised relief-valve actuation setpoints,

(3) Increased emergency core cooling (ECC) flow,

(4) Lower operating pressures,

(5) Earlier initiation of ECC systems,

(6) Heat removal through emergency condensers,

(7) Offset valve setpoints to open fewer valves per challenge,

(8) Installation of additional reliev valves with a block- or isolation-valve feature to eliminate opening of the safety/relief valves (SRVs), consistent with the ASME Code,

(9) Increasing the high steam line flow setpoint for main steam line isolation valve (MSIV) closure,

(10) Lowering the pressure setpoint for MSIV closure,

(11) Reducing the testing frequency of the MSIVs,

(12) More-stringent valve leakage criteria, and

(13) Early removal of leaking valves.

An investigation of the feasibility and contraindications of reducing challenges to the relief valves by use of the aforementioned methods should be conducted. Other methods should also be included in the feasibility study. Those changes which are shown to reduce relief-valve challenges without compromising the performance of the relief valves or other systems should be implemented. Challenges to the relief valves should be reduced substantially (by an order or magnitude).

RESPONSE

Mississippi Power & Light has participated in a BWR Owners' Group evaluation of possible ways to reduce challenges to safety/relief valves. The results of that evaluation were forwarded to the NRC in a letter from D. W. Waters to D. G. Eisenhut dated March 31, 1981. It is Mississippi Power & Light's position that further modifications to the Grand Gulf Nuclear Station would not significantly reduce the frequency of SRV events.

18.1.30.5 Report on Outages of Emergency Core-Cooling Systems Licensee Permit and Proposed Technical Specification Changes (II.K.3.17)

REQUIREMENT

Several components of the emergency core-cooling (ECC) systems are permitted by technical specifications to have substantial outage times (e.g., 72 hours for one diesel-generator; 14 days for the HPCI system). In addition, there are no cumulative outage time limitations for ECC systems. Licensees should submit a report detailing outage dates and lengths of outages for all ECC systems for the last 5 years of operation. The report should also include the causes of the outages (i.e., controller failure, spurious isolation).

RESPONSE

Mississippi Power & Light commits to reporting a summary of Emergency Core-Cooling System outages annually.

18.1.30.6 Modification of Automatic Depressurization System Logic - Feasibility for Increased Diversity for Some Event Sequences (II.K.3.18)

REQUIREMENT

The automatic depressurization system (ADS) actuation logic should be modified to eliminate the need for manual actuation to assure adequate core cooling. A feasibility and risk assessment study is required to determine the optimum approach. One possible scheme that should be considered is ADS actuation on low reactor-vessel water level provided

no high-pressure coolant injection (HPCI) or high-pressure coolant system (HPCS) flow exists and a low-pressure emergency core cooling (ECC) system is running. This logic would complement, not replace, the existing ADS actuation logic.

RESPONSE

Mississippi Power & Light has participated in a BWR Owners' Group study to simplify ADS actuation without degrading other functionally related ECCS systems. This study is complete and is presently being reviewed by General Electric for proposed modifications. Modifications to the ADS actuation logic will be documented in future FSAR revisions as they are implemented.

18.1.30.7 Restart of Core Spray and Low-Pressure, Coolant-Injection Systems (II.K.3.21)

REQUIREMENT

The core-spray and low-pressure, coolant-injection (LPCI) system flow may be stopped by the operator. These systems will not restart automatically on loss of water level if an initiation signal is still present. The core spray and LPCI system logic should be modified so that these systems will restart, if required, to assure adequate core cooling. Because this design modification affects several core-cooling modes under accident conditions, a preliminary design should be submitted for staff review and approval prior to making the actual modification.

RESPONSE

Mississippi Power & Light has endorsed the position of the BWR Owners' Group delineated in the letter from D. B. Waters to D. G. Eisenhut dated December 29, 1980. That position is that the current LPCI, LPCS, and HPCS system design is adequate and no design changes are required. However, although not required for safety considerations, a modification to the HPCS system to automate restart on low level following manual trip is planned for Grand Gulf Nuclear Station. This modification will be addressed in a subsequent FSAR amendment.

18.1.30.8 Automatic Switchover of Reactor Core Isolation Cooling System Suction - Verify Procedures and Modify Design (II.K.3.22)

REQUIREMENT

The reactor core isolation cooling (RCIC) system takes suction from the condensate storage tank with manual switchover to the suppression pool when the condensate storage tank level is low. This switchover should be made automatically. Until the automatic switchover is implemented,

licensees should verify that clear and cogent procedures exist for the manual switchover of the RCIC system suction from the condensate storage tank to the suppression pool.

RESPONSE

The RCIC System design at Grand Gulf Nuclear Station incorporates the automatic RCIC suction transfer from the Condensate Storage Tank (CST) to the Suppression Pool upon a CST low level signal or a Suppression Pool high level signal. Further discussion of the RCIC System is included in Section 5.4.6.

18.1.30.9 Confirm Adequacy of Space Cooling for High-Pressure Coolant Injection and Reactor Core Isolation Cooling Systems (II.K.3.24)

REQUIREMENT

Long-term operation of the reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) system may require space cooling to maintain the pump-room temperatures within allowable limits. Licensees should verify the acceptability of the consequences of a complete loss of alternating-current power. The RCIC and HPCI systems should be designed to withstand a complete loss of offsite alternating-current power to their support systems, including coolers, for at least 2 hours.

RESPONSE

Grand Gulf Nuclear Station utilizes safety-related pump rooms cooled by unit coolers and support systems designed to withstand the consequences of a complete loss of offsite AC power. Loss of offsite AC power results in power being supplied from the engineered safety features bus. Refer to Section 9.4.5 for a further discussion of safety-related ventilation and cooling systems.

18.1.30.10 Effect of Loss of Alternating-Current Power on Pump Seals (II.K.3.25)

REQUIREMENT

The licensees should determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating-current (ac) power for at least 2 hours. Adequacy of the seal design should be demonstrated.

RESPONSE

Mississippi Power & Light has participated in a BWR Owners' Group evaluation on the effect of loss of alternating current power on recirculation pump seals; and, has detertermined that no change in design is necessary.

Grand Gulf Nuclear Station provides a reliable source of cooling water to the recirculation pump seal coolers which can continue to operate following loss of offsite power. Even in the case of loss of cooling, followed by extreme degradation of the pump seals, the primary coolant loss, analyzed to be less than 70 gpm, will be compensated for by normal or emergency water level controls. Consequently, no hazard to the health and safety of the public will result from total loss of recirculation pump seal cooling water.

18.1.30.11 Provide Common Reference Level for Vessel Level Instrumentation (II.K.3.27)

REQUIREMENT

Different reference points of the various reactor vessel water level instruments may cause operator confusion. Therefore, all level instruments should be referenced to the same point. Either the bottom of the vessel or the top of the active fuel are reasonable reference points.

RESPONSE

Mississippi Power & Light has endorsed the position of the BWR Owners' Group delineated in the letter from Mr. D. G. Waters to Mr. D. G. Eisenhut dated December 29, 1980. That position is basically that "...identification of a common water level reference is not vital to ensure safe reactor operation; and consequently, no modification of the current control room water level instrumentation is required on the basis of plant safety considerations." However, the following actions are being taken:

1) Incorporate a common water level reference (to instrument zero) on the Safety Parameter Display System (see Item I.D.2).

2) Place a mimic on control room panel 1H13-P601 above the reactor level recorders and display instruments to show references to the different water levels.

18.1.30.12 Verify Qualification of Accumulators on Automatic Depressurization System Valves (II.K.3.28)

REQUIREMENT

Safety analysis reports claim that air or nitrogen accumulators for the automatic depressurization system (ADS) valves are provided with sufficient capacity to cycle the valves open five times at design pressures. GE has also stated that the emergency core cooling (ECC) systems are designed to withstand a hostile environment and still perform their function for 100 days following an accident. Licensee should verify that the accumulators on the ADS valves meet these requirements, even considering normal leakage. If this cannot be demonstrated, the licensee must show that the accumulator design is still acceptable.

RESPONSE

Mississippi Power & Light is participating with the BWR Owners' Group in performing a generic evaluation in response to this requirement. The results of this evaluation are scheduled to be available for review in July 1981; at which time, MP&L will prepare a response to this item.

18.1.30.13 Revised Small-Break Loss-of-Coolant-Accident Methods to Show Compliance with 10 CFR Part 50, Appendix K (II.K.3.30)

REQUIREMENT

The analysis methods used by nuclear steam supply system (NSSS) vendors and/or fuel suppliers for small-break loss-of-coolant accident (LOCA) analysis for compliance with Appendix K to 10 CFR Part 50 should be revised, documented, and submitted for NRC approval. The revisions should account for comparisons with experimental data, including data from the LOFT Test and Semiscale Test facilities.

RESPONSE

Mississippi Power & Light understands that the NRC and General Electric are working to resolve staff concerns about small break LOCA models for BWRs.

18.1.30.14 Plant-Specific Calculations to Show Compliance with 10 CFR Part 50.46 (II.K.3.31)

REQUIREMENT

Plant-specific calculations using NRC-approved models for small-break loss-of-coolant accidents (LOCAs) as described in item II.K.3.30 to show compliance with 10 CFR 50.46 should be submitted for NRC approval by all licensees.

RESPONSE

Mississippi Power & Light will perform plant-specific calculations following NRC approval of LOCA model revisions required by item II.K.3.30.

18.1.30.15 Evaluation of Anticipated Transients with Single Failure to Verify No Fuel Failure (II.K.3.44)

REQUIREMENT

For anticipated transients combined with the worst single failure and assuming proper operator actions, licensees should demonstrate that the core remains covered or provide analysis to show that no significant fuel damage results from core uncovery. Transients which result from a stuck-open relief valve should be included in this category.

RESPONSE

Mississippi Power & Light is a participant in the BWR Owners' Group generic evaluation of Item II.K.3.44 which addresses the issue of adequate core cooling for transients with a single failure. The results of this evaluation, which applies to Grand Gulf, was submitted to the NRC in a letter from Mr. D. B. Waters, Owners' Group Chairman, to Mr. D. G. Eisenhut, dated December 29, 1980.

18.1.30.16 Evaluation of Depressurization with Other Than Automatic Depressurization System (II.K.3.45)

REQUIREMENT

Analyses to support depressurization modes other than full actuation of the automatic depressurization system (ADS) (e.g., early blowdown with one or two safety relief valves (SRVs) should be provided. Slower depressurization would reduce the possibility of exceeding vessel integrity limits by rapid cooldown.

RESPONSE

Mississippi Power & Light is a participant in the BWR Owners' Group generic evaluation of Item II.K.3.45 which addresses the issue of alternate modes of depressurization other than full actuation of the ADS. The results of this evaluation, which applies to Grand Gulf, was submitted to the NRC in a letter from Mr. D. B. Waters, Owners' Group Chairman, to Mr. D. G. Eisenhut, dated December 29, 1980.

18.1.30.17 Michelson's Concerns (II.K.3.46)

REQUIREMENT

General Electric should provide a response to the Michelson concerns as they relate to boiling water reactors.

RESPONSE

The General Electric Company has responded to the questions posed by Mr. Michelson in the letter from R. Bushholz to D. Ross dated Februrary 21, 1980. This response is applicable to the Grand Gulf Nuclear Station.

18.1.31 Emergency Preparedness - Short Term (III.A.1.1)

REQUIREMENT

Comply with Appendix E, "Emergency Facilities," to 10 CFR Part 50, Regulatory Guide 1.101, "Emergency Planning for Nuclear Power Plants," and for the offsite plans, meet essential elements of NUREG-75/111 or have a favorable finding from FEMA.

RESPONSE

Mississippi Power & Light submitted Revision 1 of the Grand Gulf Nuclear Station Emergency Response Plan by letter AECM-81/83 dated May 14, 1981. This response satisfies the requirements of this item.

18.1.32 Upgrade Emergency Support Facilities (III.A.1.2)

Mississippi Power & Light has provided the plan for upgrading and providing emergency response facilities at Grand Gulf Nuclear Station in letter AECM-81/52, dated February 12, 1981. Additionally, letter AECM-81/25 of April 8, 1981 responded to NUREG-0696 with a conceptual description of the Emergency Operations Facility.

18.1.33 Improving Licensee Emergency Preparedness - Long Term (III.A.2)

REQUIREMENT

Each nuclear facility shall upgrade its emergency plans to provide reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency. Specific criteria to meet this requirement is delineated in NUREG-0654 (FEMA-REP-1), "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparation in Support of Nuclear Power Plants."

RESPONSE

Mississippi Power & Light submitted Revision 1 of the Grand Gulf Nuclear Station Radiological Emergency Response Plan by letter AECM-81/23 dated May 14, 1981. This response included elements of NUREG-0654, Revision 1, Appendix 2.

Additionally, letter AECM-81/103 of April 10, 1981 provided information concerning the meteorological requirements of NUREG-0654 and Regulatory Guide 1.23, "Meteorological Measurements Programs in Support of Nuclear Power Plants."

18.1.34 Integrity of Systems Outside Containment Likely to Contain Radioactive Material for Pressurized-Water Reactors and Boiling-Water Reactors (III.D.1.1)

REQUIREMENT

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate leak reduction

(a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment.

(b) Measure actual leakage rates with system in operation and report them to the NRC.

(2) Continuing Leak Reduction -- Establish and implement a program of preventive maintenance to reduce leakage to as-los-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

RESPONSE

Specific procedures for implementing a leakage reduction program at Grand Gulf Nuclear Station are being developed. The following is a summary of the intended leak reduction program for systems outside containment that could contain highly radioactive fluids during an accident.

The Leak Reduction Program is primarily dependent on collecting leakage in the sumps located in the room in which the system is located. An accurate measurement of sump level will be taken, then the system will be operated for a sufficient time to detect abnormal leakage rates and the level will be taken again. Total system leak rate can then be calculated. When abnormal leak rates are detected, the source of the leak will be determined and corrected. Additionally, the system will be walked down to visually inspect for signs or leakage.

In cases where potential leakage will not be collected in a sump such as leakage of gas or steam, or where components are not serviced by a sump, the following alternative methods will be used.

Water Leakage

Water leakage will be detected by direct observation where practical. When ALARA or other considerations dictate, leakage will be collected. Observable leakage past vent and drain valves will be eliminated. Valve packing leakage will be minimized.

Steam Leakage

Steam leakage from the RCIC system will be identified by having an Iodine and particulate airborne radioactivity sample taken while the system is operating. Abnormal activity will require further investigation. This method of leak detection cannot be used prior to power operations.

Gas Leakage

Gas leakage will be detected by liquid soap application on each joint. Any detected leakage will be eliminated.

Each identified system will be checked for leakage as part of the Surveillance Test Procedure. Initial leak test results will be supplied as soon as available. Note that steam leak check results will not be quantifiable in gpm and they will not be available until power operation is achieved.

The following systems are included, to the extent indicated, in the program.

Reactor Core Isolation Cooling System

Entire System outside containment containment steam or water except drain line to main condenser.

Residual Heat Removal System

Entire System outside containment containing steam or water except line to Liquid Radwaste System and some headers that are isolated by manual valves.

High Pressure Core Spray System

Entire System outside containment.

Low Pressure Core Spray System

Entire System outside containment.

Combustible Gas Control System

Hydrogen Analyzers only.

Suppression Pool Make-Up System

Suppressioıı Pool Level detection portion of the system.

Feedwater Leakage Control System

Entire system.

Post-Accident Sampling System when installed

Entire system.

Systems containing radioactive materials which are excluded from the program follow with the justification for exclusion.

A. MSIV Leakage Control System

This system draws leakage from the main steam lines between the MSIV's and the outboard shut-off valve and exhaust into the Auxiliary Building so that the Leakage will be processed by the Standby Gas Treatment System (SGTS). The MSIV Leakage Control System operates at a negative pressure, hence leakage would be into the system and of no concern.

B. Standby Gas Treatment System

The SGTS functions to collect and process post LOCA Containment Leakage. Leakage out of the SGTS is into regions served by the system and would not increase the radioactivity levels existing in the Auxiliary Building during post-LOCA operation.

C. Reactor Water Clean-Up (RWCU)

The system is not required to function during or immediately following an accident and is isolated from post accident fluids. Possible system usage would be under controlled conditions wuch that the system could be prepared for such usage in the long term post accident situation.

D. Suppression Pool Cleanup System

See justification for C.

E. Off-Gas System

See justification for C.

F. Liquid and Solid Radwaste System

See justification for C.

Gaseous systems to be tested include the Hydrogen Analyzers and associated piping in the Combustible Gas Control Systems. Each item

will be tested while in normal running condition by checking each mechanical joint with Liquid Soap.

The program to reduce potential release paths will be provided in GGNS reply to IE Bulletin 80-10.

18.1.35 Improved Inplant Iodine Instrumentation Under Accident Conditions (III.D.3.3)

REQUIREMENT

(1) Each licensee shall provide equipment and associated training and procedures for accurately determing the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

(2) Each applicant for a fuel-loading license to be issued prior to January 1, 1981 shall provide the equipment, training, and procedures necessary to accurately determine the presence of airborne radioiodine in areas within the plant where plant personnel may be present during an accident.

RESPONSE

Inplant iodine monitoring at Grand Gulf Nuclear Station will be accomplished by use of continuous air monitors and portable low volume samplers with subsequent laboratory analysis of filter media. A further discussion of inplant iodine monitoring under accident conditions is contained in Section 12.5.2.2.5.

18.1.36 Control Room Habitability Requirements (III.D.3.4)

REQUIREMENT

In accordance with Task Action Plan item III.D.3.4 and control room habitability, licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19, "Control Room," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50).

RESPONSE

The control room design at Grand Gulf Nuclear Station meets the habitability requirements of GDC 19 of 10 CFR 50 Appendix A and the guidelines of Regulatory Guides 1.78 and 1.95. Section 6.4 provides a complete description of the control room HVAC system layout and functional design that provides for protection of the control room from radioactive and toxic gases.

Additional information required for the Control Room Habitability Evaluation:

1. Control Room Mode of Operations: See descriptions of the modes of operation for the Control Room HVAC Systems in Sections 6.4.2 and 9.4.1.

2. Control Room Characteristics:

   a. Control Room air volume: See Section 6.4.2.2.

   b. Control Room emergency zone: See Section 6.4.2.1.

   c. Control Room ventilation system schematic with normal and emergency air flow rates: See Figure 6.5-1.

   d. Infiltration leakage rate: See Section 6.4.2.3; and, Tables 6.4-1 and 15.6-13.

   e. HEPA filter and charcoal adsorber efficiencies: See Section 6.5.1.4.1 and Table 15.6-13.

   f. Closest distance between containment and air intake: Approximately 43 feet; however, this is not unobstructed distance as the containment is completely enclosed by a secondary containment consisting of the auxiliary building and enclosure building (see Figure 1.2-4).

   g. Layout of Control Room, air intakes, Containment Building, and chlorine or other chemical storage facility with dimensions: See Figure 2.2-5 and Table 2.2-6.

   h. Control Room shielding, including radiation streaming from penetrations, doors, ducts, stairways, etc.: See Section 6.4.2.5; and, Tables 15.6-12, 15.6-13 and 15.6-14.

   i. Automatic isolation capability - damper closing time, damper leakage, and area: Butterfly valves with leakage requirements as specified by MSS-SP-67 for Type 1 valves are utilized as automatic isolation dampers for the Control Room HVAC systems, see Table 18.1-2 for a summary of the characteristics of these dampers.

   j. Chlorine detectors or toxic gas: See Sections 6.4.1.1e, 6.4.2.2, 6.4.4.2, 7.3.1.1.10, 9.4.1.1.1e, 9.4.1.3, and 9.4.1.5.

   k. Self-contained breathing apparatus availability: Will be provided.

   l. Bottled air supply: See Section 6.4.2.6.

   m. Emergency food and potable water supply: Supplies for five men for five days.

n. Control Room personnel capacity: See Section 6.4.1.1.

o. Potassium iodide drug supply: Will be provided.

TABLE 18.1-1
TMI CONTAINMENT ISOLATION EVALUATION

Containment Penetration Number	System Name	Classification	Automatic Isolation (see Table 6.2-44 for details)	Discussion (as necessary)
4	Fuel Pool Cooling & Cleanup Transfer Tube	Non-essential	No (Locked Closed)	N/A
5	Nuclear Boiler - Main Steam Lines	Essential	Yes	Penetration is within the reacto coolant pressure boundary but is not required to be open for mitigation of accidents.
6	Nuclear Boiler - Main Steam Lines	Essential	Yes	Same as 5, above.
7	Nuclear Boiler - Main Steam Lines	Essential	Yes	Same as 5, above.
8	Nuclear Boiler - Main Steam Lines	Essential	Yes	Same as 5, above.
9	Nuclear Boiler - Feedwater Inlet	Essential	Reverse flow for check valves; remote - manual for motor-operated shutoff valve.	Same as 5, above. Also, feedwater inlet is a potential source of makeup to the reactor vessel i available.
10	Nuclear Boiler - Feedwater Inlet	Essential	Reverse flow for check valves; remote - manual for motor-operated shutoff valve.	Same as 9, above.
11	RHR Pump - "A" Suction	Essential	No	Emergency Core Cooling
12	RHR Pump - "B" Suction	Essential	No	Same as 11, abov
13	RHR Pump - "C" Suction	Essential	No	Same as 11, abov
14	RHR Reactor Shutdown Cooling Suction	Essential	Yes	Same as 5, above.

TABLE 18.1-1
TMI CONTAINMENT ISOLATION EVALUATION

Containment Penetration Number	System Name	Classification	Automatic Isolation (see Table 6.2-44 for details)	Discussion (as necessary)
17	Steam Supply to RHR and RCIC Turbine	Essential	Yes	Same as 5, above.
18	RHR to Head Spray	Essential	Yes	Same as 5, above.
19	Nuclear Boiler - Main Steam Drains	Essential	Yes	Same as 5, above.
20	RHR Heat Exchanger "A" to LPCI	Essential	No	Emergency Core Cooling System (ECCS)
21	RHR Heat Exchanger "B" to LPCI	Essential	No	Emergency Core Cooling System
22	RHR Pump "C" to LPCI	Essential	No	Emergency Core Cooling System
23	RHR - "A" Pump Test and Minimum Flow Line to Suppression Pool	Essential	No	ECCS - Suppressio Pool Cooling Return Line
24	RHR - "C" Pump Test and Minimum Flow Line to Suppression Pool	Essential	Yes for 14" connection, no for 4" pump minimum flow line	Test line only fo ECCS pump.
25	HPCS Pump Suction	Essential	No	Emergency Core Cooling System
26	HPCS Pump Discharge	Essential	No	Emergency Core Cooling System
27	HPCS Test Line	Essential	Yes for 12" connection, No for 4" pump minimum flow line	Test line only fc ECCS Pump
28	RCIC Pump Suction	Essential	No	RCIC provides makeup to RPV in the event of loss of all AC power.
29	RCIC Turbine Exhaust	Essential	No	Same as 28, above

TABLE 18.1-1
TMI CONTAINMENT ISOLATION EVALUATION

Containment Penetration Number	System Name	Classification	Automatic Isolation (see Table 6.2-44 for details)	Discussion (as necessary)
30	LPCS Pump Suction	Essential	No	Emergency Core Cooling System
31	LPCS Pump Discharge	Essential	No	Emergency Core Cooling System
32	LPCS Test Line	Essential	Yes for 14" connection, No for 4" pump minimum flow line	Test line only fo ECCS pump.
33	CRD Pump Discharge	Beneficial	No (Remote Manual Only)	CRD provides potential source of high pressure makeup to reactor pressure vessel. Class 1E instrumentation i provided to monitor functiona integrity of piping inside containment. Lin can be isolated b the operator from control room base on status of CRD system.
34	Containment Purge and Ventilation Air Supply	Non-essential	Yes	N/A
35	Containment Purge and Ventilation Air Exhaust	Non-essential	Yes	N/A
36	Plant Service Water Return	Beneficial	No (Remote - Manual only)	Plant service water provides coolant for drywell coolers. These coolers are not required to mitigate the consequences of accidents but are helpful in maintaining drywell temperatures

TABLE 18.1-1
TMI CONTAINMENT ISOLATION EVALUATION

Containment Penetration Number	System Name	Classification	Automatic Isolation (see Table 6.2-44 for details)	Discussion (as necessary)
				during and after transients. The Plant Service Water System also provides water fc fire protection i the drywell. Class 1E instrumentation is provided to monitor functiona integrity of piping inside the containment (and drywell). This line can be isolated by the plant operator from the control room based on the status of the Plant Service Water System.
37	Plant Service Water Supply	Beneficial	No (Remote - manual only)	Same as 36, above
38	Plant Chilled Water Supply	Non-essential	Yes	N/A
39	Plant Chilled Water Return	Non-essential	Yes	N/A
40	Integrated Leak Rate Test Connection	Non-essential	No (Blank Flange)	N/A
41	Service Air Supply	Non-essential	Yes	N/A
42	Instrument Air Supply	Non-essential	Yes	N/A
43	RWCU to Main Condenser	Non-essential	Yes	N/A
44	Component Cooling Water Supply	Beneficial	No	CCW provides cool ing water to recirc pump seal coolers.

TABLE 18.1-1
TMI CONTAINMENT ISOLATION EVALUATION

Containment Penetration Number	System Name	Classification	Automatic Isolation (see Table 6.2-44 for details)	Discussion (as necessary)
45	Component Cooling Water Return	Beneficial	No	CCW provides cooling water to recirc pump seal coolers.
46	RCIC Pump Minimum Flow Bypass	Essential	No	RCIC provides makeup to RPV in the event of loss of all AC power.
48	RHR Heat Exchanger "B" Relief Valve Vent Header to Suppression Pool	Essential	No	This normally closed line is of the Emergency Core Cooling System which allows venting of non-condensible gases from the heat exchanger.
49	RWCU Backwash Transfer Pump to Spent Resin Tank	Non-essential	Yes	N/A
50	Drywell & Containment Equipment Drain Sump Pump Discharge	Non-essential	Yes	N/A
51	Drywell & Containment Floor Drain Sump Pump Discharge	Non-essential	Yes	N/A
54	To & From Refueling Water Storage Tank - Upper Containment Pool	Non-essential	No (Locked Closed Manual Valves)	N/A
56	Condensate Supply to Containment	Non-essential	Yes	N/A
57	To Upper Containment Pool from Fuel Pool Cooling & Cleanup System	Non-essential	Yes	N/A

TABLE 18.1-1
TMI CONTAINMENT ISOLATON EVALUATION

Containment Penetration Number	System Name	Classification	Automatic Isolation (see Table 6.2-44 for details)	Discussion (as necessary)
58	From Upper Containment Pool to Fuel Pool Drain Tank	Non-essential	Yes	N/A
60	Auxiliary Building Drains Pumpback to Suppression Pool	Non-essential	Yes	N/A
65	Combustible Gas Control Containment Purge (Outside Air Supply)	Non-essential	Yes	N/A
66	From Purge Radiation Air Detection System to Containment Exhaust Charcoal Filter Train	Non-essential	Yes	N/A
67	RHR Pump "B" Test Line to Suppression Pool	Essential	No	ECCS - Suppressic Pool Cooling Return Line
69	Refueling Water Transfer Pump Suction	Non-essential	Yes	N/A
70	Instrument Air Supply to ADS Receivers	Non-essential	Yes	N/A
71A	LPCS Relief Valve Vent Header to Suppression Pool	Essential	No (Relief Valve Discharge Line)	ECCS piping over-pressure protection discharge tc suppression pool.
71B	RHR "C" Relief Valve and Post-Accident Samply Return to Suppression Pool	Essential	No (Relief valve discharge line and locked closed MOV)	Same as 71A, above, and is al required for pos accident samplius return to suppression pool.
73	RHR Shutdown Vent Header to Suppression Pool	Non-essential	No (Relief valve discharge line)	This portion of the RHR system is not required for mitigation of accidents. However, relief valves provide reliable isolatic of the containment.

TABLE 18.1-1
TMI CONTAINMENT ISOLATION EVALUATION

Containment Penetration Number	System Name	Classification	Automatic Isolation (see Table 6.2-44 for details)	Discussion (as necessary
75	RCIC Turbine Exhaust Vacuum Breaker	Essential	Yes (on high dry-well pressure or RCIC line break only)	Provides vacuum breaker operation to prevent induction of water into turbine exhaust piping once turbine is shutdown.
76B	RHR Shutdown Suction Relief Valve Discharge	Nonessential	No (Relief valve discharge line)	Same as 73, above
77	RHR Heat Exchanger "A" Relief Valve Vent Header to Suppression Pool	Essential	No	Same as 48, above
81	Post-Accident Sample	Essential	No	This line is required to be available to allc the post-accident sampling from the recirc system jet pump lines.
82	Integrated Leak Rate Test Connection	Non-essential	No (Blind Flange)	N/A
83	RWCU Line from Regen. Heat Exchanger to Feedwater	Non-essential	Yes	N/A
84	Chemical Waste Sump Pump Discharge	Non-essential	Yes	N/A
85	Suppression Pool Cleanup Return	Non-essential	Yes	N/A
86	Demineralized Water Supply to Containment	Non-essential	Yes	N/A
87	RWCU Pump Discharge	Non-essential	Yes	N/A
88	RWCU Pump Discharge	Non-essential	Yes	N/A

TABLE 18.1-1
TMI CONTAINMENT ISOLATION EVALUATION

tem Name	Classification	Automatic Isolation (see Table 6.2-44 for details)	Discussion (as necessary)	Discussion (as necessary
ndby Service Water pply "A"	Essential	No	Provides essential cooling water to safety related equipment located inside containment.	Allows monitoring of containment pressure during normal, transient and accident conditions; also allows for post-accident sampling of the containmen atmosphere.
ndby Service Water turn "A"	Essential	No	Same as 89, above.	
ndby Service Water turn "B"	Essential	No	Same as 89, above.	N/A
ndby Service Water pply "B"	Essential	No	Same as 89, above.	Allow monitoring of suppression pool level during all normal, transient, and accident conditions.
ywell & Containment essure Instruments	Essential	No	Monitor pressure inside containment and drywell during normal, transient, and accident conditions.	
st-Accident Sample	Essential	No	Allows for post-accident sampling of the reactor recirc system.	
ywell & Containment drogen Analyzer mple & Return Lines	Essential	No	Monitor hydrogen concentration in the drywell and in the containment during normal, transient, and accident conditions.	
ywell Fission Product nitor Sample & Return	Essential	No	Allow monitoring of airborne fission products in the drywell atmosphere during all normal, transient and accident conditions.	

TABLE 18.1-2
CONTROL ROOM ISOLATION DAMPER CHARACTERISTICS

Equipment No.	Size	Leakage	Maximum Closing Time
QSZ51F001	24"	0	4 Sec.
QSZ51F002	24"	0	4 Sec.
QSZ51F003	18"	0	4 Sec.
QSZ51F004	18"	0	4 Sec.
QSZ51F007	20"	0	4 Sec.
QSZ51F010	18"	0	4 Sec.
QSZ51F011	18"	0	4 Sec.
QSZ51F016	20"	0	4 Sec.

ATTACHMENT TO AECM-81/153

FSAR CHANGE PAGES REQUIRED TO SUPPORT
REVIEW OF CHAPTER 18.

d. Administrative Controls (~~4~~ 1 ~~weeks~~ week)

The administrative controls segment consists of a detailed presentation of the Grand Gulf Nuclear Station Administrative Procedures that pertain to the administrative activities necessary to operate the unit. Such topics as technical specifications, control and handling of radioactive materials, protective tagging, etc., will be covered.

Insert A

~~e. General Operating Procedures (1 week)~~

~~The General Operating Procedures portion will contain the presentation of the Grand Gulf Integrated Operating Instructions, System Operating Instructions, Alarm Response Instructions, Emergency Plan and Site Security Plan.~~ 47

~~f. Transient and Accident Analysis and Emergency Procedures (3 weeks)~~

~~This portion will give a detailed presentation of the transient and accident analysis section of the Grand Gulf Final Safety Analysis Report. The Emergency Procedures will be covered concurrent with transient and accident analyses so that an overall understanding can be obtained. Combined with these two topics will be an instructional period devoted to accessing and interpreting information supplied from the process computer.~~

~~g.~~ f. STA Simulator Training (2½ weeks)

This 2½-week portion is designed to familiarize the STAs with a fundamental understanding of system and plant operation in a control room atmosphere. The time is split between actual control room operation and classroom presentations.

The STA Training Program is intended to be a short-term plan which ensures that technical expertise is available to the Shift Supervisor for matters dealing with accident/transient response of the GGNS plants. The long-range plan is to certify the Shift Superintendents to the level of STA at which time there may no longer be a need for a separate STA.

13.2.1.2.11 Training for Technical Support Personnel

The plant staff training section shall provide periodic training to all technical support personnel on the nuclear plant engineering and nuclear project services staff. This training should be planned to aid these persons in performing their normal job functions and to ensure that they are prepared to adequately support the plant staff in an emergency. Although 48

INSERT A TO 13.2.1.2.10

e ~~5.~~ Plant Operations Course (4 weeks)

The Plant Operations Course will contain the presentation of the Grand Gulf Integrated Operating Instructions, System Operating Instructions, Alarm Response Instructions, Emergency Plan, Site Security Plan, and Off Normal Event Procedures.

In addition, the Plant Operations Course will give a detailed presentation of the transient and accident analysis section of the Grand Gulf Final Safety Analysis Report. The Emergency Procedures will be covered concurrent with transient and accident analyses so that an overall understanding can be obtained.

GG
FSAR

TABLE OF CONTENTS

12.6 DESIGN REVIEW OF PLANT SHIELDING

TABLE OF CONTENTS (Cont.)

LIST OF TABLES (Cont.)

TABLE OF CONTENTS (Cont.)

LIST OF FIGURES (Cont.)

331.13

12.6-6

Post-Accident Radiation Zones
Unit 1, Plan at EL 208'-10"

12.6 DESIGN REVIEW OF PLANT SHIELDING

12.6.1 INTRODUCTION

NUREG-0737, Item II.B.2, "Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May Be Used In Postaccident Operation," identified the requirement that a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials be performed.

The radiation and shielding-design review was performed to ~~define~~ identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during postaccident operations of these systems. Additionally, the review results ensure that adequate access to vital areas and protection of safety related equipment is provided through the use of design changes, additional shielding, or administrative control changes.

12.6.2 SYSTEMS IDENTIFIED FOR SHIELDING REVIEW

Systems outside containment which will or may have to function during a serious transient or accident and which may contain highly radioactive materials can be classified into two categories as follows:

a. Those systems required for plant shutdown or mitigation of accident consequences which are also expected to contain highly radioactive materials.

b. Those other systems directly connected to the reactor coolant system, suppression pool or containment atmosphere which, while neither designed to nor expected to contain highly radioactive material, are postulated to contain such material.

¶ The systems containing radioactive fluid, identified for review of plant shielding and environmental qualifications are:

a. Portions of the RHR system during the following accident modes;

1. LPCI Mode - the portions used to inject the suppression pool water.

2. Shutdown Cooling Mode - Portions to provide "normal" residual heat removal service (although not expected to be used).

3. Steam Condensing Mode - Portions used for condensing steam in the RPV vapor dome. (although not expected to be used).

4. Suppression Pool Cooling Mode - Portions used to recirculate the suppression pool water for cooling.

5. Containment Spray Mode - Portion used to recirculate suppression pool water for containment spray.

b. Portions of the low pressure core spray (LPCS) system used to recirculate the suppression pool water.

c. Portions of the high pressure core spray (HPCS) system used to recirculate the suppression pool water.

d. Portions of the reactor core isolation cooling (RCIC) systems as follows:

1. Steam supply piping and turbine exhaust piping.

2. Pump suction and discharge piping during the steam condensing mode of RPV cooldown (although not expected to be used).

3. RCIC pump suction from suppression pool.

e. Feedwater leakage control system

f. MSIV leakage control system and SGTS

g. Post-accident sampling system

h ~~8~~. Portions of combustible gas control system associated with H_2 analyzers.

i ~~9~~. Portions of suppression pool makeup system associated with RHR/LPCS sealing water for suppression pool level instrumentation.

j ~~7~~. Portion of the drains system located in auxiliary building required to accomplish radwaste pump-back function (system under design development).

12.6.3 SHIELDING REVIEW METHODOLOGY

12.6.3.1 Source Term

The fission product release data used in the shielding review analysis was obtained from General Electric for a 1 MW_t BWR-6 with a total burn-up of 1095 MWd. This data was then adjusted by the maximum design power load of 4025 MW_t to obtain the appropriate fission product inventory for the Grand Gulf core. This fission product inventory was then partitioned into three categories of sources as described below.

12.6.3.2 Release Fractions

For those systems which contain pressurized reactor coolant, 100% of the noble gases, 50% of the iodines and 1% of the particulates were assumed to be present with a dilution volume equal to the pressure vessel liquid volume (11,085 ft^3). Systems containing recirculating liquid were assumed to have 50% of the iodine and 1% of the particulates diluted

by the suppression pool volume plus the pressure vessel liquid volume (146,320 ft^3, total). All airborne sources were assumed to contain 100% of the noble gases and 25% of the iodines. The dilution volume for contained airborne sources was selected two ways. If the system contains drywell air, only the drywell volume was used; otherwise, the entire volume of the containment plus drywell was used for dilution. Instantaneous mixing between the drywell and containment was assumed for all containment through-wall and through-penetration doses; this assumption maximized these contributions.

12.6.3.3 Post-Accident Radiation Zone Maps

Using the sources of 12.6.3.2, dose rate vs. distance curves were ~~dal~~ developed for containment wall and penetrations contributions using the QAD-CG code; and, for contained source contributions from various pipe sizes using point kernal methodology. For contained sources, infinite pipe length was conservatively assumed. The contributions from all sources were then summed to develop post accident radiation zone maps, Figures 12.6-1 thru 12.6-6.

12.6.4 Identification of Areas Outside Containment for Review

Areas outside containment which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident are identified in Table 12.6-1.

12.6.5 Integrated Personnel Exposures

later

Table 12.6 -1

AREAS OUTSIDE OF CONTAINMENT FOR
REVIEW OF PLANT SHIELDING

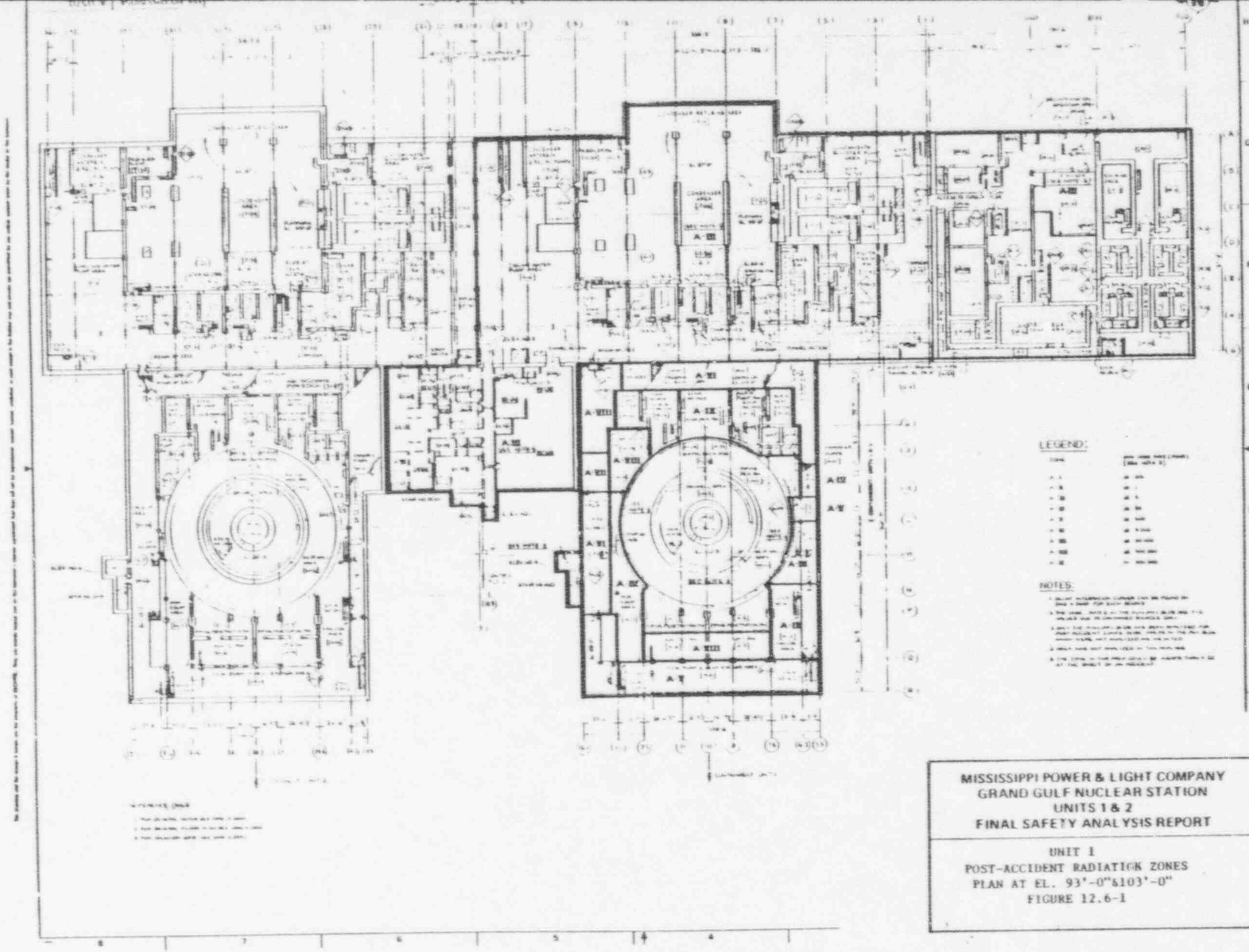
BUILDING	ELEVATION	AREA
Auxiliary Bldg	93'-0"	RHR A, B & C Equipment Rooms RCIC equipment room HPCS equipment room LPCS equipment room ECCS instrument rack areas Floor and equipment transfer tanks and pumps rooms Refuel water transfer pump room Control rod drive pumps and filter area
Auxiliary Bldg	119'-0"	RHR equipment rooms ESF switchgear rooms MSIV leakage control system equipment area Postaccident sample room

Table 12.6-1 (cont.)

Auxiliary Bldg	135'-0"	Standby gas treatment equipment room RHR equipment rooms BOP computer multiplexi[illegible] & isolation panels areas CRD RPV temperature recorder panel are[illegible]
Auxiliary Bldg	166'-0"	Containment hydrogen sample rack areas Radiation monitoring rack areas Containment exhaust charcoal filter room Instrument air boost[illegible] compressors area
Auxiliary Bldg	184'-0"	Vicinity of blowout shaft area
Auxiliary Bldg	208'-10"	Enclosure bldg fans
Control Bldg	93'-0"	Health physics are[illegible]

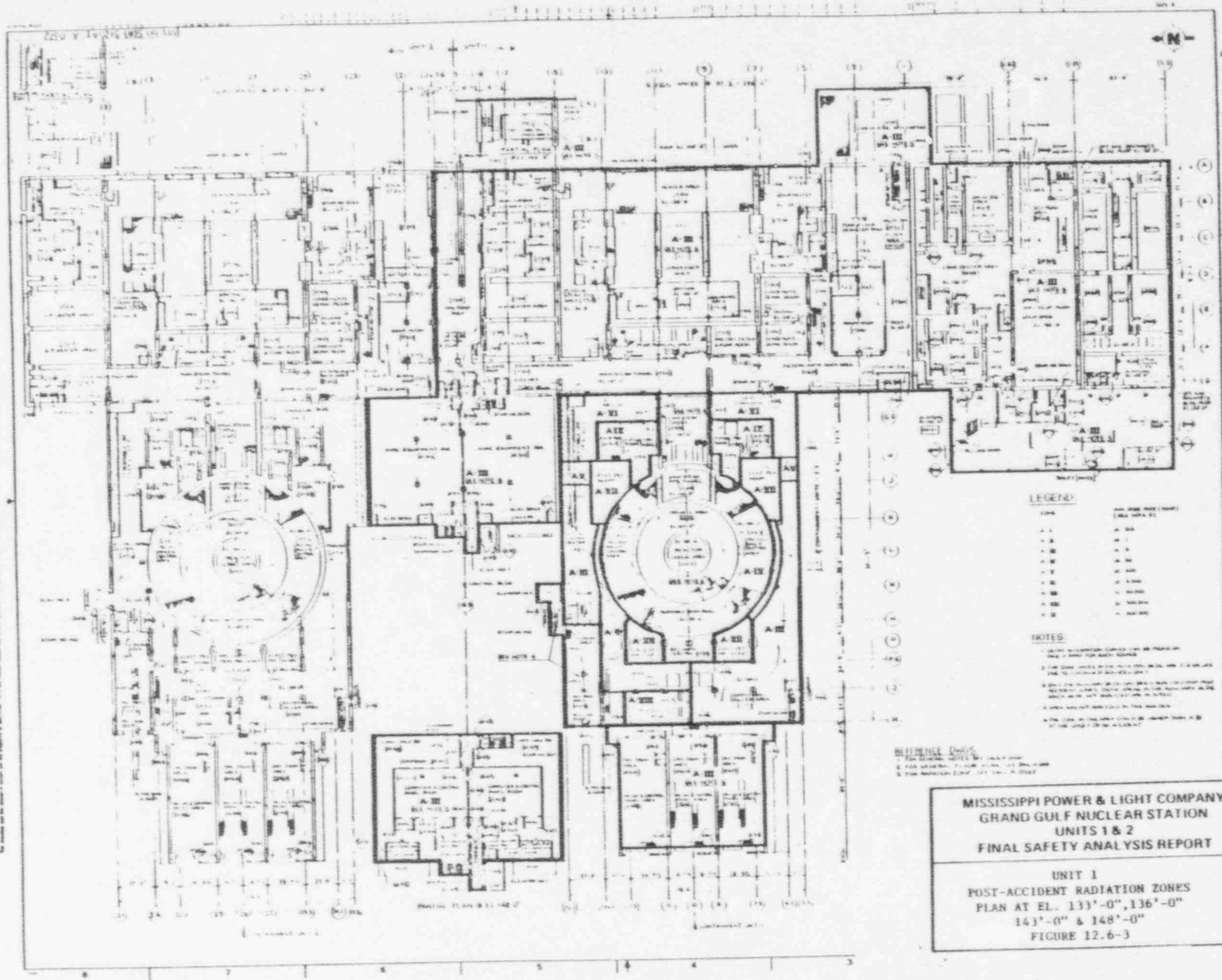
Table 12.6-1
(cont.)

Control Bldg	111'-0"	Remote shutdown panel area
Control Bldg	148'-0"	Lower cable spreading room
Control Bldg	166'-0"	Main control room
Control Bldg	177'-0"	Technical support center
Control Bldg	189'-0"	Upper cable spreading room
Turbine Bldg	—	General access to all areas
Radwaste Bldg	118'-0"	Radioactive chemistry laboratory
Diesel Generator Bldg	—	General access to all areas



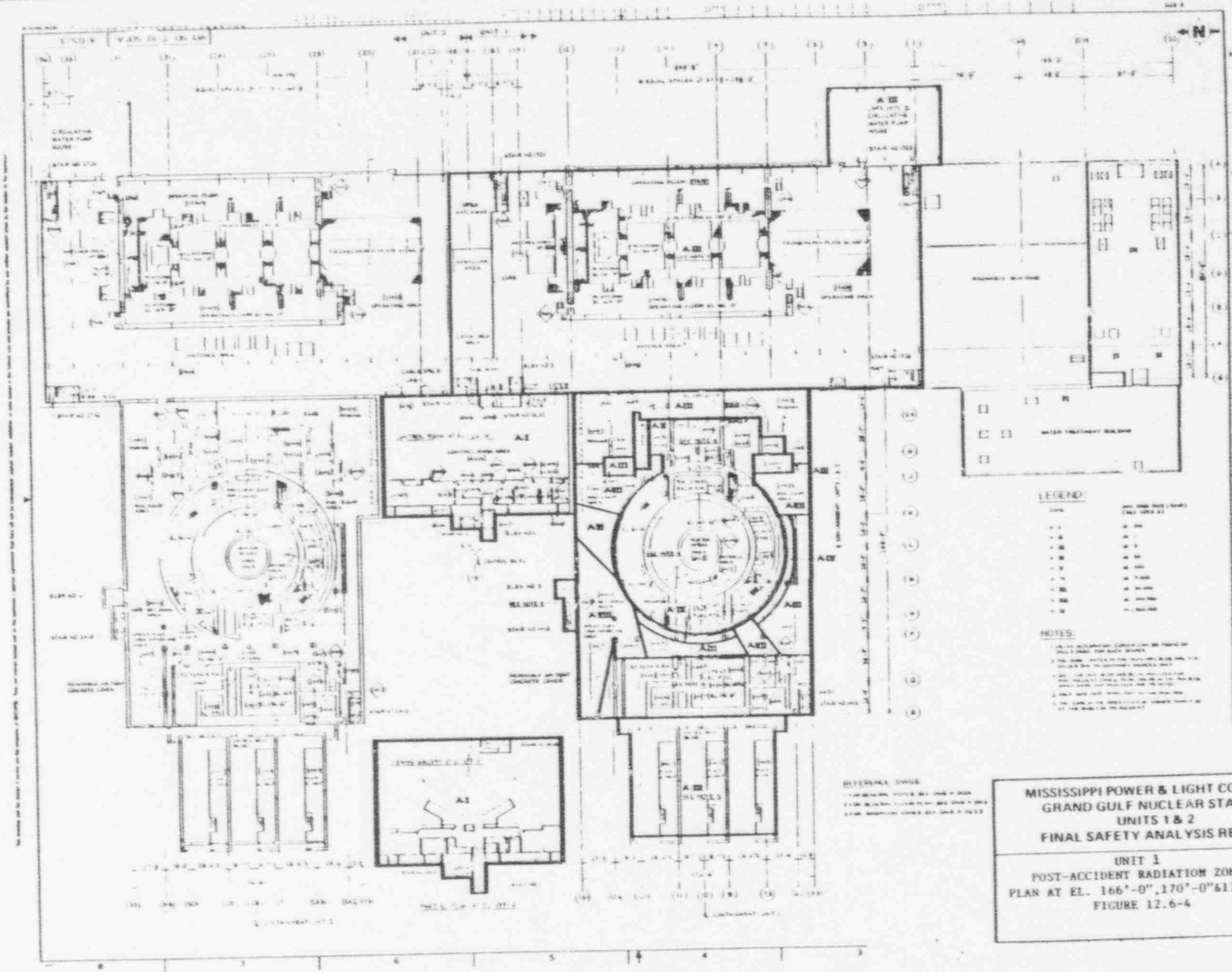
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

UNIT 1
POST-ACCIDENT RADIATIGN ZONES
PLAN AT EL. 93'-0"&103'-0"
FIGURE 12.6-1



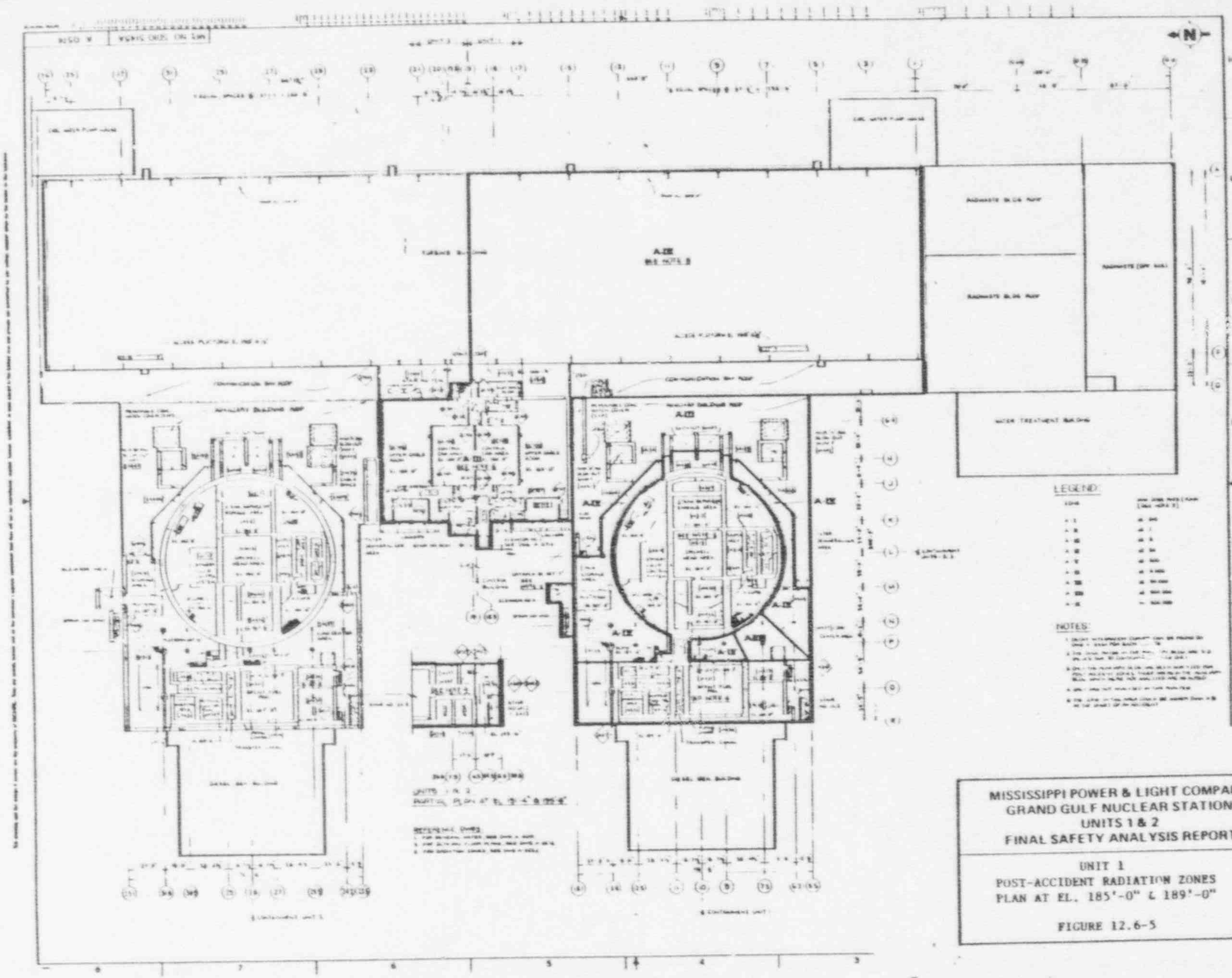
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

UNIT 1
POST-ACCIDENT RADIATION ZONES
PLAN AT EL. 133'-0",136'-0"
143'-0" & 148'-0"
FIGURE 12.6-3



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

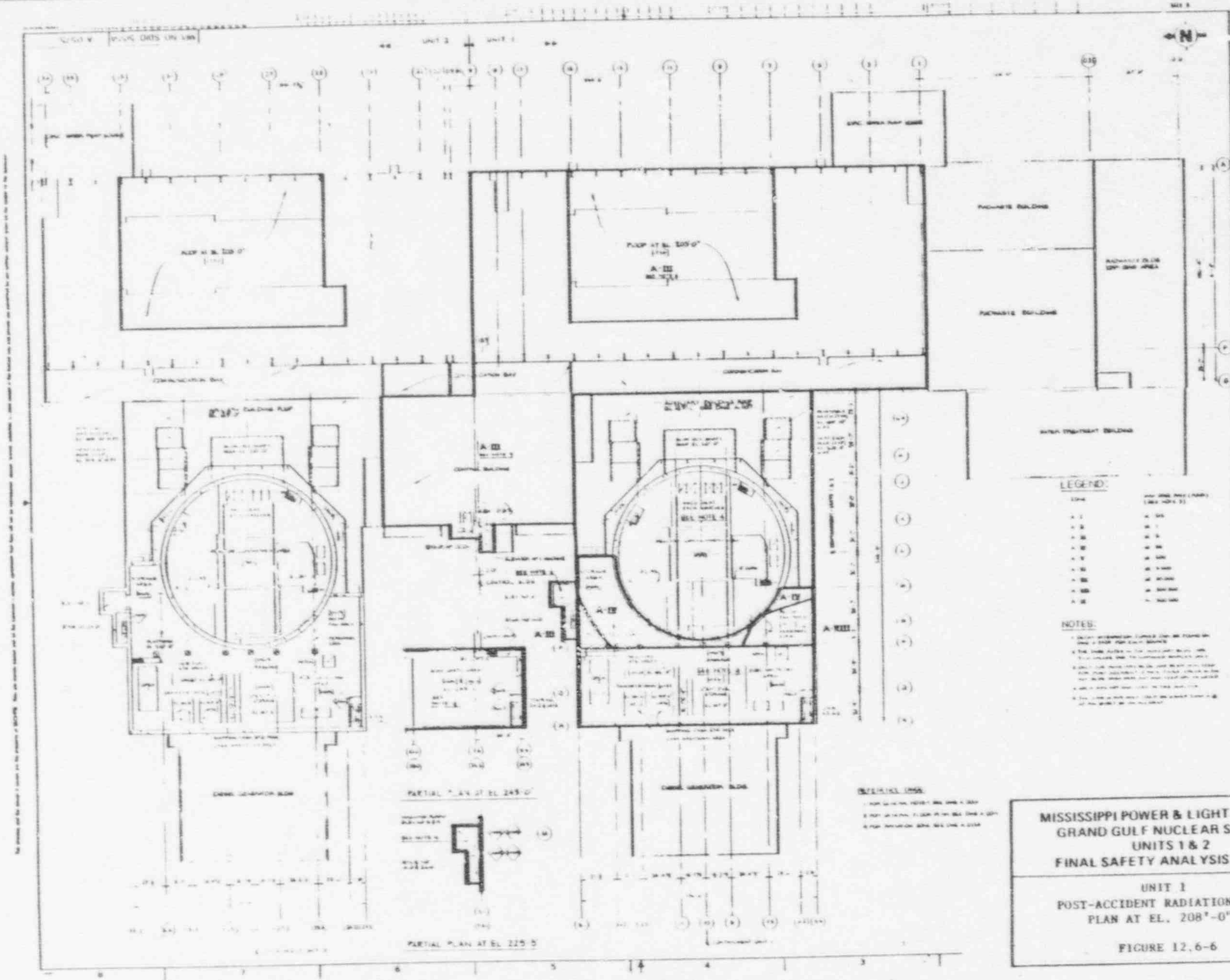
UNIT 1
POST-ACCIDENT RADIATION ZONES
PLAN AT EL. 166'-0",170'-0"&177'-0"
FIGURE 12.6-4



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

UNIT 1
POST-ACCIDENT RADIATION ZONES
PLAN AT EL. 185'-0" & 189'-0"

FIGURE 12.6-5



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

UNIT 1
POST-ACCIDENT RADIATION ZONES
PLAN AT EL. 208'-0"

FIGURE 12.6-6

7.0 Instrumentation and Control Systems

7.7 - Control Systems Not Required for Safety

7.7.1 Description

ADD → K. Process Sampling System (PSS)

ADD THE FOLLOWING NEW SECTIONS:

7.7.1.11 Process Sampling System - Instrumentation and Controls

7.7.1.11.1 System Identification

7.7.1.11.1.1 General

The purpose of the process sampling system instrumentation and controls is to collect representative liquid and gas samples for analysis and to provide analytical information required to monitor plant and equipment performance and changes to operating parameters.

7.7.1.11.1.2 Classification

This is a power generation system and is classified as not related to safety.

7.7.1.11.2 Power Sources

The PSS instrumentation and controls are powered from non-essential busses.

7.7.1.11.3 Equipment Design

7.7.1.11.3.1 General

The process sampling system is described in subsection 9.3.2.

7.7.1.11.3.2 Testability

Since the process sampling system is usually in service during plant operation, satisfactory performance is demonstrated without the need for any special inspection ~~and~~ or testing beyond that specified in the manufacturer's instructions.

7.7.1.11.4 Operational Considerations

7.7.1.11.4.1 General Information

The process sampling system is manually operated at grab sample panels located throughout the plant. The sampling panels are designed to minimize contamination and radiation at the sample station. Appropriate shielding and area radiation monitors will minimize radiation effects.

7.7.1.11.4.2 Post Accident Sampling Station

The post accident portion of the process sampling system is designed with the following operational considerations:

a. The system is capable of obtaining and analyzing reactor coolant and containment atmosphere samples within 3 hours from the time a decision is made to take a sample.

b. Facilities are provided onsite to perform the analysis described in subsection 9.3.2.2.4, within the 3 hour time frame established above.

c. Reactor coolant and containment atmosphere sampling during post accident conditions does not require an isolated auxiliary system to be placed in operation in order to use the sampling system.

d. Chloride analysis as well as the analyses described in subsection 9.3.2.2.4 can be performed using in-line monitoring.

e. The post accident sampling station is designed to provide adequate radiation protection so that it is possible for an operator to obtain and analyze a sample without radiation exposures exceeding the criteria of GDC 19.

f. The system is designed for in-line monitoring with grab sampling as a backup. The equipment provided for backup sampling is capable of providing at least one sample per day for 7 days following onset of the accident and at least one sample per week until the accident condition no longer exists.

g. The systems radiological and chemical analysis capability includes provisions to identify and quantify the isotopes of the nuclide categories of concern, to levels corresponding to the source terms given in Regulatory Guides 1.3 and 1.7.

h. The gamma detection system allows the monitoring of the reactor coolant activity over a range of 10^{-7} to 10^{1} Ci/cc and the containment atmosphere activity over a range of 10^{-9} to 10^{-1} Ci/cc during normal and accident conditions.

i. The sample station is designed to restrict backround levels of radiation such that the sample analysis provides results with an acceptably small error.

j. The post accident sampling system instrumentation provides adequate ranges, accuracies, and sensitivities to allow the operator to obtain pertinent data in order to describe the radiological and chemical status of the reactor coolant systems.

K. Reactor coolant sample lines are of a diameter such that the rupture of a sample line will limit reactor coolant loss.

7.7.1.4.4.3 Operator Information

The most important liquid and gas samples are analyzed continuously by the PSS with alarms in the makeup water treatment area when measured values go beyond normal limits. Analog signals are indicated on recorders and/or processed in the computer.

7.0 Instrumentation and Control Systems
7.7 Control Systems Not Required for Safety
7.7.2 Analysis

ADD THE FOLLOWING NEW SECTIONS:

7.7.2.11 Process Sampling System - Instrumentation and Controls

7.7.2.11.1 Conformance to General Functional Requirements

The process sampling system is not a safety-related system. Therefore, the instrumentation supplied is for the plant equipment protection and for operator information only. None of this instrumentation is required to initiate or control any engineered safeguards or safety systems.

7.7.2.11.2 Conformance to Specific Regulatory Requirements

There are no specific regulatory requirements imposed on this system.

TABLE 7.1-1 (Cont.)

System	Designer	Supplier
Control Systems Not Required for Safety		
Reactor Vessel Instrumentation	NSSS	NSSS
Rod Control and Information System-Rod Movement Control	NSSS	NSSS
Recirculation Flow Control System	NSSS	NSSS
Feedwater Control System	NSSS	Partially-NSSS
Pressure Control and Turbine Generator System	Partially-NSSS	Non-NSSS
Neutron Monitoring System Traverse Incore Probe (TIP)	NSSS	NSSS
NSSS Process Computer System	NSSS	NSSS
Reactor Water Cleanup System (RWCU)	NSSS	Partially-NSSS
Gaseous Radwaste System	NSSS	NSSS
Auxiliary Building Pressure Control System	Non-NSSS	Non-NSSS
Process Sampling System (PSS)	Non-NSSS	Non-NSSS

TABLE 7.1-2

Instrumentation and Controls (System)	Plants Applying for or Having Construction Permit or Operating License	Similarity of Design
(31) Recirculation Flow Control System	Zimmer-1	Identical
(32) Feedwater Control System	Zimmer-1	Identical
(33) Pressure Control and Turbine Generator System	None	New design
(34) NSSS Process Computer	Vermont Yankee and subsequent plants	See Note 7
(35) Reactor Water Cleanup System	Zimmer-1, LaSalle	See Note 8
(36) Gaseous Radwaste System	Hanford-2	See Note 15
(37) Auxiliary Building Pressure Control System	None	New design
(38) ATWS Mitigation Capability	None	Later - design not determined
(39) Process Sampling System	None	New design

Q31.83 Q31.71 Q31.80 Q31.83 Q31.83

TABLE 7.1-3

IDENTIFICATION OF SAFETY CRITERIA [3]

SHEET 1 OF 2

Column key:
- Reactor Protection System: 1 Reactor Protection System
- Engineered Safety Features Systems: 2 Containment and Reactor Vessel Isolation Control; 3 Emergency Core Cooling; 4 Main Steamline Isolation Valve Leakage Control; 5 Feedwater Leakage Control; 6 Combustible Gas Control; 7 RHR–Containment Spray Cooling System; 8 Standby Gas Treatment; 9 Suppression Pool Makeup; 10 Control Room Atmospheric Control and Isolation
- Auxiliary Supporting Systems: 11 Standby Service Water; 12 Standby Power System (SEE SUBSECTION 8.3.1.2); 13 HVAC for ESF Areas; 14 Diesel Generator Auxiliaries
- Systems Required for Safe Shutdown: 15 Remote Shutdown; 16 Reactor Core Isolation Cooling; 17 Standby Liquid Control; 18 Reactor Shutdown Cooling
- 19 Safety Related Display Instrumentation
- All Other Systems Required for Safety: 20 Refueling Interlocks; 21 Process Radiation Monitoring; 22 High Pressure/Low Pressure Interlock Protection; 23 Leak Detection; 24 Neutron Monitoring; 25 Rod Pattern Control System; 26 Recirculation Pump Trip; 27 Spent Fuel Pool Cooling & Cleanup; 28 Component Cooling Water Fuel Pool Cooling; 29 Suppression Pool Temperature Monitoring; 30 Auxiliary Building Isolation Control
- Control Systems Not Required for Safety: 31 Reactor Vessel Instrumentation; 32 Recirculation Flow Control; 33 Feedwater Control; 34 Pressure Control and Turbine Generator; 35 Process Computer; 36 Reactor Water Cleanup; 37 Gaseous Radwaste; 38 Auxiliary Building Pressure Control; 39 Rod Control & Information; 40 Post Accident Sampling

Criteria	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36	37	38	39	40
10 CFR 50.34	X	X	X		X	X	X	X	X	X	X		X	X	X								X					X	X	X				X				X		
10 CFR 50.36	X	X	X		X	X	X	X	X	X	X		X	X	X								X					X	X	X										
10 CFR 50.36a	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X		
GDC 1	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X										
GDC 2	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X										
GDC 3	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X										
GDC 4	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X										
GDC 5	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X										
GDC 10	X	X					X																X																	
GDC 12	X																																							
GDC 13	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X		X		X		X	X	X	X		X	X	X	X			X						X
GDC 15	X																																							
GDC 19	X	X	X	X	X	X	X	X	X	X	X		X	X	X			X	X				X	X		X		X	X	X				X				X		X
GDC 20	X	X	X			X	X	X	X	X	X		X	X	X			X			X		X	X		X			X	X										
GDC 21	X	X	X	X	X	X	X	X	X	X	X		X	X	X			X			X		X	X		X		X	X	X										
GDC 22	X	X	X	X	X	X	X	X	X	X	X		X	X	X			X			X		X	X		X		X	X	X										
GDC 23	X	X	X	X	X	X	X	X	X	X	X		X	X	X			X			X		X	X		X		X	X	X										
GDC 24	X	X	X	X	X	X	X	X	X	X	X		X	X	X			X			X		X	X	X	X		X	X	X	X	X		X				X		
GDC 25	X																																							
GDC 26																	X								X														X	
GDC 27																									X															
GDC 28																								X																
GDC 29	X	X	X		X	X	X	X	X	X	X		X	X	X		X	X			X			X		X		X		X										
GDC 30																							X																	
GDC 33			X				X									X							X																	
GDC 34		X					X				X				X			X	X				X																	
GDC 35			X						X														X																	
GDC 37			X						X										X																					
GDC 38							X				X																													
GDC 40							X												X																					
GDC 41						X																																		
GDC 43						X																																		
GDC 44											X																	X												
GDC 46											X																	X												
GDC 50																																								
GDC 54						X					X												X																	
GDC 55																																								
GDC 56						X																																		
GDC 57																																								
GDC 60																																					X			
GDC 61																											X													
GDC 63																											X													
GDC 64		X									X										X							X												X
IEEE 279-1971	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X		X	X	X	X	X	X		X	X	X										
IEEE 308-1971		X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X							X														
IEEE 317-1972	X	X	X			X	X		X							X	X	X	X		X		X	X	X	X			X		X	X				X				
IEEE 323-1971	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X		X		X	X	X	X		X	X	X										
IEEE 334-1971[1]						X																																		
IEEE 336-1971	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X			X		X	X	X		X	X	X										
IEEE 338-1971	X	X	X	X	X	X		X	X	X	X		X	X	X	X	X	X	X			X	X	X	X	X		X	X	X										
IEEE 344-1971[1]	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X		X		X	X	X	X		X	X	X										
IEEE 379-1972	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X		X	X	X	X	X	X		X	X	X										
IEEE 384-1974	X	X	X		X	X	X	X	X	X	X		X	X	X	X	X	X	X		X				X			X	X	X										

(1) LATER REVISIONS OF THESE DOCUMENTS HAVE BEEN ISSUED. (IEEE 323-1974, IEEE 344-... WISE, BUT GE HAS NOT DETERMINED A PROGRAM FOR COMPLIANCE TO THEM. IT IS EX...

(2) REGULATORY GUIDE 1.89 IS AN ENDORSEMENT OF IEEE 323-1974. GE HAS NOT YET ESTABLISHED A POSITION ON THE GUIDE BUT WILL PROVIDE SOME DEGREE OF COM...

(3) IDENTIFICATION OF SAFETY CRITERIA

TABLE 7.1-3

IDENTIFICATION OF SAFETY CRITERIA (Cont.) (3)

SHEET 2 OF 2

	REACTOR PROTECTION SYSTEM	ENGINEERED SAFETY FEATURES SYSTEMS									AUXILIARY SUPPORTING SYSTEMS				SYSTEMS REQUIRED FOR SAFE SHUTDOWN				SAFETY RELATED DISPLAY INSTRUMENTATION	ALL OTHER SYSTEMS REQUIRED FOR SAFETY												CONTROL SYSTEMS NOT REQUIRED FOR SAFETY									
		CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL	EMERGENCY CORE COOLING	MAIN STEAMLINE ISOLATION VALVE LEAKAGE CONTROL	FEEDWATER LEAKAGE CONTROL	COMBUSTIBLE GAS CONTROL	RHR-CONTAINMENT SPRAY COOLING SYSTEM	STANDBY GAS TREATMENT	SUPPRESSION POOL MAKEUP	CONTROL ROOM ATMOSPHERIC CONTROL AND ISOLATION	STANDBY SERVICE WATER	STANDBY POWER SYSTEM	HVAC FOR ESF AREAS	DIESEL GENERATOR AUXILIARIES	REMOTE SHUTDOWN	REACTOR CORE ISOLATION COOLING	STANDBY LIQUID CONTROL	REACTOR SHUTDOWN COOLING		REFUELING INTERLOCKS	PROCESS RADIATION MONITORING	HIGH PRESSURE/LOW PRESSURE INTERLOCK PROTECTION	LEAK DETECTION	NEUTRON MONITORING	ROD PATTERN CONTROL SYSTEM	RECIRCULATION PUMP TRIP	SPENT FUEL POOL COOLING & CLEANUP	COMPONENT COOLING WATER	FUEL POOL COOLING	SUPPRESSION POOL TEMPERATURE MONITORING	AUXILIARY BUILDING ISOLATION CONTROL	REACTOR VESSEL INSTRUMENTATION	RECIRCULATION FLOW CONTROL	FEEDWATER CONTROL	PRESSURE CONTROL AND TURBINE GENERATOR	PROCESS COMPUTER	REACTOR WATER CLEANUP	GASEOUS RADWASTE	AUXILIARY BUILDING PRESSURE CONTROL	ROD CONTROL & INFORMATION	PROCESS SAMPLING
NUCLEAR REGULATORY COMMISSION REGULATORY GUIDES																																									
1.6			X		X	X	X	X	X	X	X		X	X	X	X		X	X										X	X	X										
1.7						X																																			
1.11	X	X	X			X	X		X						X	X			X																						
1.22	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X			X	X	X		X	X			X	X	X										
1.29	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X		X		X	X	X				X	X	X										
1.30	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X				X						X	X	X										
1.32			X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X										X	X	X										
1.40						X																																			
1.45																							X																		
1.47	X	X	X	X	X	X	X	X	X	X	X			X		X	X	X	X		X		X	X	X	X			X	X	X										
1.53	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X		X	X	X	X		X			X	X	X										
1.56																																					X				
1.62	X	X	X	X		X	X	X	X	X	X					X	X																								
1.63	X	X	X			X	X		X		X					X	X	X	X		X		X	X	X	X				X		X	X				X				
1.68																								X																	
1.69	X				X	X		X	X	X	X		X	X	X				X	X	X	X			X				X	X	X	X	X	X	X	X	X	X	X		
1.73		X	X			X	X		X		X																														
1.75	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X		X	X			X			X	X	X	X										
1.80																																									
1.89(2)	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X	X	X	X		X		X	X	X				X	X	X										
1.95										X																															
1.96				X																																					
1.100		X	X	X	X	X	X	X	X	X	X					X		X											X		X										
BRANCH TECHNICAL POSITIONS – EICSB																																									
1																																									
3			X		X																																				
4																																									
6																																									
9																																									
10	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X		X	X		X		X	X	X				X	X	X										
12																																									
13																																									
14																																									
16																																									
17																																									
18																																									
19																																									
20																																									
21	X	X	X	X	X	X	X	X	X	X	X			X		X	X	X	X		X		X	X	X				X	X	X										
22	X	X	X	X	X	X	X	X	X	X	X		X	X	X	X		X					X		X				X	X	X										
23																			X																						
24																																									
25			X																																						
26	X																																								
27																																									

SEE SUBSECTION 8.3.1.2

9.3.2 Process Sampling System

9.3.2.1 Design Bases

9.3.2.1.1 Safety Design Bases

a. The seismic design and quality group classifications of sample lines and their components conform to the classification of the system to which they are connected up to and including the second isolation valve.

b. ~~Sampling points located inside the containment terminate at the containment sampling station. No sampling line penetrates the containment wall.~~

INSERT "A"

c. All sampling lines have the process isolation valves located as close as practical to the process taps.

d. The sampling panels are designed to minimize contamination and radiation at the sample station. Appropriate shielding and area radiation monitors minimize radiation effects.

9.3.2.1.2 Power Generation Design Bases

The process sampling system (PSS) collects representative liquid and gas samples for analysis and provides the analytical information required to monitor plant and equipment performance and changes to operating parameters.

The process sampling system is designed to function during all plant operational modes under individual system requirements. Design guidelines related to PSS capabilities, obtaining representative samples and safety are described in the following paragraphs and Table 9.3-3.

9.3.2.2 System Description

9.3.2.2.1 General Description

The process sampling system provides sampling of all principal fluid process streams associated with plant operation.

The process sampling system consists of:

a. Permanently installed sampling nozzles and sample lines.

b. Sampling panels with analyzers and associated sampling equipment.

c. Provisions for local grab sampling.

INSERT "A" (9.3.2.1.1.b)

Sample points located inside the containment terminate at the containment sampling station, and do not penetrate the containment wall, with the exception of the sample points for the post accident sampling station. The post accident samples located inside the containment will be run as close as possible to the outside of the containment wall and will be lead shielded.

9.3.2.2.2 Sampled Process Streams and Analyzed Parameters

The process streams to be sampled are shown on P&IDs with the sample point symbols. The SE symbol is used for Remote Sample Points which are connected with remote sampling panels or analyzers. The SX symbol designates Local Sample Points provided for local grab sampling, see Figures 1.8-1 and 1.8-2. Figures 9.3-5 through -8 show the most important sampling panels and condenser leakage detection sampling. The sample points and the local analyzers are shown on P&ID of individual systems. For sampling of radioactive gases see Section 11.5. Table 9.3-3 provides a list of sample points, associated P&ID figures, and analyzed parameters.

9.3.2.2.3 Provisions for Obtaining Representative Samples

The following provisions are incorporated into the PSS:

a. Where practicable, a sample takeoff connection is located in a turbulent flow zone, where fluid streams are well mixed, after a minimum straight run of 3 pipe diameters of process pipe. (Where physically possible, a straight run of 10 pipe diameters is preferred).

b. The connection is made at the side of horizontal process pipe.

c. Sampling nozzles designed for insertion into the streams are provided for process pipes 2-1/2 inches and larger, unless the process or fluid conditions dictate otherwise.

d. Sampling lines are sized to maintain turbulent flow and to minimize purge time. Routing is as short and straight as possible. Tubing with large radius bends are used to avoid traps and dead legs.

e. Sampling nozzles, lines, and associated valves and fittings are fabricated from stainless steel material.

f. Heat tracing of sampling lines is provided where necessary to prevent crystalization or solidification of contents.

g. Sampling equipment is designed for flushing and blowndown in order to remove sediment deposits, air, and gas pockets. Provisions are made to purge sample lines and to reduce plateout or precipitation in sample lines.

h. Provisions are made to sample the bulk volume of tanks. The standby liquid control system storage tank will be sampled from the top opening so that any low points and potential sediment traps can be avoided.

9.3.2.2.4 Sampling Panels

Different process conditions, water quality, and analyzing equipment require special treatment of individual sample streams. These specific requirements are incorporated in the design of the process sampling system whose P&IDs are shown in Figures 9.3-5 through 9.3-8B.

Figure 9.3-5 shows the reactor water sample station and feedwater corrosion product monitor supplied by GE as part of the NSSS. Further discussion related to sampling and analysis of reactor coolant and the coolant chemistry requirements is provided in subsection 5.2.3.2.2.

Figures 9.3-6 through 9.3-8A show the individual sampling trains inside the non-NSSS supplied sampling panels. A typical high pressure and high temperature sample passes through the inlet panel shut-off valve, through the filter, is rough cooled, and pressure reduced to about 25 psig. The second cooler provides the final temperature conditioning to 77 F before the sample enters the conductivity or pH analyzers. Special manual valves are provided to take grab samples for laboratory analysis.

The sample stream can be monitored by temperature and pressure indicators. Flow through each analyzer is adjustable by means of the sample flowmeter. The train is connected to a common header through another flowmeter for continuous blowdown. Samples entering the sampling panel at low temperature (below 140 F) require no rough cooling. Pressure reduction equipment is also provided for initial sample pressure. For convenience, the sampling panels are divided into several sections, each taking care of special functions:

a. Pressure and temperature conditioning section.

b Grab sampling section with a sink which can be flushed with demineralized water, sample hood, and other equipment for protection against radioactive contamination.

c. Analyzer and monitoring section.

A reclamation header is provided in the turbine building sampling station for all clean water continuous samples to minimize waste processing requirements.

— — - INSERT "B" - - — —

INSERT "B" (add to end of 9.3.2.2.4)

Figure 9.3-8B shows the post accident sampling station. Samples are admitted into the system by means of valves which are controlled remotely. Demineralized water is used to cool the incoming samples and to purge the system. Purge control is accomplished remotely. Following passage through the detector subsystem, the coolant samples are cooled with a tube-in-shell type, stainless steel sample cooler. The sample noncondensible gases are analyzed by separating the liquid in a steam trap assembly.

The system is designed for in-line monitoring with grab sampling as a backup. The in-line portion consists of gamma detectors for gas and water, instrumentation for isotopic analysis, measurement of hydrogen concentration in the noncondensible gases, determination of the ratio of liquid to noncondensible gases, and analyzers for dissolved oxygen, conductivity, pH, and chloride ion in the reactor coolant. (refer to Table 9.3.3.5). The grab sample portion is capable of obtaining any of the following samples:

a. Pressurized, undiluted reactor coolant
b. Pressurized, undiluted containment atmosphere
c. Depressurized, diluted reactor coolant separated liquid
d. Depressurized, undiluted reactor coolant separated gases

All grab samples utilize sample cylinders, having double shut-off, quick disconnects and remotely operated valves.

INSERT "B"

The post accident sampling station is provided with return lines for disposal of the samples; liquid samples are returned to the suppression pool through the RHR system and the gas samples are returned to the drywell. The ventilation exhaust from the sampling station is filtered with a charcoal adsorber and HEPA filters.

9.3.2.3 Safety Evaluation

a. The sample nozzles and lines are designed, fabricated, installed, and tested in accordance with the requirements of the process lines from which the samples are taken.

b. The reactor water sample line (recirculation system) penetrating the drywell wall has two motor-operated isolation valves (one inside and one outside the drywell), which close automatically on an isolation signal from the Containment and Reactor Vessel Isolation Control System. This line is classified as ASME Code, Section III, Class 2.

c. Reactor coolant to the Reactor Sample Station is sampled with a GE designed sample nozzle which has an 1/8 inch diameter port hole facing into the flow stream. This type of sampling nozzle is suitable for obtaining a representative sample and also provides a restriction to limit reactor coolant loss from a rupture of the sample line.

d. Sample nozzles are stress analyzed for the most severe process conditions to avoid failure.

e. Pressure reduction valves and other devices (pressure regulators and safety relief valves) are provided for protection of operators and/or equipment (refer to Figures 9.3-5 through 9.3-8B.

f. Reactor water and main steam sample lines are of sufficient length to permit decay of short lived radio-nuclides in order to protect sampling personnel.

g. All sample lines connected to seismic Category I systems are analyzed as seismic Category I lines up to and including the second isolation valve; main process pipe code classification is applicable. Sampling lines from the second isolation valve to the panel/analyzer are in conformance with ANSI B31.1, Power Piping Code.

h. All sample lines have provisions for purging and draining sample streams to an appropriate waste treatment system or to the systems of their origin.

9.3.2.4 Tests and Inspection

The sample nozzles and associated piping, tubing, fittings, and valves are tested and inspected in accordance with the requirements of the main process pipes from which the samples are taken.

TABLE 9.3-3

LIST OF SAMPLING POINTS

Sample	Sample Point	Figure	Parameters
a.	Containment Building Sampling Station at El. 184 - 6		
B33-D014	Reactor water recirculation inlet	5.4-2	C, O_2, G
G33-D014	Reactor water cleanup filter demineralizers influent	5.4-21	C, O_2, G
G33-D015A	Reactor water cleanup filter demineralizer A, effluent	5.4-21	C, G
G33-D015B	Reactor water cleanup filter demineralizer B, effluent	5.4-21	C, G
C11-D004	CRD drive water filters discharge	4.6-8	C, O_2, G
b.	Turbine Building Sample Conditioning and Analyzing Panel at El. 93 - 0		
N11-N060	Main steam line C	10.3-3	G
N21-N007	Reactor feed pump discharge	10.4-13	C, pH, O_2, G
N19-N420	Condensate pumps discharge	10.4-10	C, Na, G
N22-N450	Cond demin effluent A	10.4-9	C, Na, G

TABLE 9.3-3 (Cont.)

Sample		Sample Point	Figure	Parameters
N22-N451		Cond demin effluent B	10.4-9	C, Na, G
N22-N452		Cond demin effluent C	10.4-9	C, Na, G
N22-N453		Cond demin effluent D	10.4-9	C, Na, G
N22-N454		Cond demin effluent E	10.4-9	C, Na, G
N22-N455		Cond demin effluent F	10.4-9	C, Na, G
N22-N456		Cond demin effluent G	10.4-9	C, Na, G
N22-N457		Cond demin effluent H	10.4-9	C, Na, G
N22-N466		Cond demin combined effluent	10.4-9	C, Na, O_2, G, Turb
	1*	LP condenser shell tube break detector A	9.3-7	C, Na
	2*	LP condenser shell tube break detector B	9.3-7	C, Na, G
	3*	IP condenser shell tube break detector A	9.3-7	C, Na, G
	4*	IP condenser shell tube break detector B	9.3-7	C, Na, G

*Selectable sampling points using common sodium and conductivity analyzers.

TABLE 9.3-3 (Cont.)

Sample		Sample Point	Figure	Parameters
	5*	HP condenser shell tube break detector A	9.3-7	C, Na, G
	6*	HP condenser shell tube break detector B	9.3-7	C, Na, G

c. Radwaste Building Sampling Panel Adjacent to Water Inventory Control Station at El. 118-0

Sample	Sample Point	Figure	Parameters
SG17-N450	Equipment drain collector tanks pump discharge	11.2-1	G
SG17-N451	Waste surge tanks pump discharge	11.2-1	G
SG17-N452	Equipment drain filter discharge	11.2-1	G
SG17-N453	Equipment drain demineralizer effluent	11.2-1	G
SG17-N454	Equipment drain sample tank discharge	11.2-1	G
SG17-N455	Floor drain collector tank pump discharge	11.2-2	G
SG17-N456	Floor drain filter discharge	11.2-2	G
SG17-N457	Floor drain demireralizer effluent	11.2-2	G

*Selectable sampling points using common sodium and conductivity analyzers.

TABLE 9.3-3 (Cont.)

Sample	Sample Point	Figure	Parameters
SG17-N458	Floor drain sample tanks discharge	11.2-2	G
SG17-N460	Misc chemical waste receiver tank pump discharge	11.2-3	G
SG17-N459A	Regeneration sol receiving tank A pump discharge	11.2-3	G
SG17-N459B	Regeneration sol receiving tank B pump discharge	11.2-3	G
SG17-N463	Distillate sample pump discharge	11.2-4	G
SG17-N464A	Evaporator bottoms pump A discharge	11.2-4	G
SG17-N464B	Evaporator bottoms pump B discharge	11.2-4	G
d.	Feedwater Corrosion Product Monitor, Turbine Building, at El. 133-0		
N21-N007	Reactor feed pump discharge	10.4-13	Turb, S
e.	Standby Liquid Control System		
1	SLC Storage Tank	9.3-24	G (Borate concentration)
f.	Circulating Water System		
N71-N422A	HP cond out A	10.4-6	Cl_2, G
N71-N422B	HP cond out B	10.4-6	Cl_2, G

TABLE 9.3-3 (Cont.)

Sample	Sample Point	Figure	Parameters
N71-N453A	CWP A disch	10.4-5	C, pH, G
N71-N453B	CWP B disch	10.4-5	C, pH, G
N71-N440	CW makeup	10.4-5	C, G
g.	Standby Service Water System		
P41-N033A	SSW pump A disch	9.2-1	C, pH, G
P41-N033B	SSW pump B disch	9.2-1	C, pH, G
P41-N034A	SSW cooling twr A rtn	9.2-1	G, R, T
P41-N034B	SSW cooling twr B rtn	9.2-1	G, R, T
P41-N032A	SSW loop A radn supl	9.2-2	G, R, T
P41-N032B	SSW loop B radn supl	9.2-2	G, R, T
h.	Extraction Steam System		
N36-N400A	Fdwtr htr 3A	10.3-4	Steam quality (
N36-N400B	Fdwtr htr 3B	10.3-4	Steam quality (
N36-N400C	Fdwtr htr 3C	10.3-4	Steam quality (
i.	Residual Heat Removal System		
E12-N100A	RHR loop A	5.4-17	G
E12-N100B	RHR loop B	5.4-16	G
j.	Plant Service Water System		
P4[illegible]-N037	PSW to CWS	9.2-22	Cl_2, G, R, T

TABLE 9.3-3 (Cont.)

Sample	Sample Point	Figure	Parameters
k.	Turbine Building Cooling Water System		
P43-N052	TBCW pumps disch	9.2-24	G
l.	Plant Chilled Water System		
P71-N031	Pri pumps disch	9.2-17	G
P71-N032	T.B. sec pumps disch	9.2-17	G
P71-N033	Hot wtr tk disch	9.2-17	G
m.	Component Cooling Water System		
P42-N450	CCW pump disch	9.2-9	G
P42-N451	CCW radn supl	9.2-9	G
n.	Fuel Pool Cooling and Cleanup System		
G41-D001	Fuel pool H/X disch	9.1-26	G, R
G41-D002	FPCC filter dem's disch	9.1-26	G, R
G46-N300A	FP filter dem D001 disch	9.1-28	C, G, R
G46-N300B	FP filter dem D002 disch	9.1-28	C, G, R
o.	Suppression Pool Cleanup System		
1P60-N450	Cond precoat filtr inlet	9.3-6	G
p.	Condensate Cleanup System		
1N22-N458	Resin sep & cation regen tank drain	10.4-9	G
1N22-N459	Anion regen tank outlet drain	10.4-9	G

TABLE 9.3-3 (Cont.)

Sample	Sample Point	Figure	Parameters
1N22-N460	Resin mix & stor tank drain	10.4-9	G
1N22-N461	Post strainer back flush	10.4-9	G
1N22-N462	Acid mixing tee effluent	10.4-9	G
1N22-N463	Caustic mixing tee effluent	10.4-9	G
1N22-N464	Rinse & bckwsh wtr supply	10.4-9	G
1N22-N465	Rinse recycle to condenser	10.4-9	G
1N22-N467	URC feed water	10.4-9	G
1N22-N468	URC waste outlet	10.4-9	G
1N22-N469	Slurry from URC eductor	10.4-9	G
1N22-N470	Precoat filt re-circ pump A DS	10.4-9	G
1N22-N471	Precoat filt re-circ pump B DS	10.4-9	G
1N22-N472	Precoat filt re-circ pump C DS	10.4-9	G
q.	Make Up Water Treatment System	9.2-11 and 9.2-12	
1	Cation D001A exch sample S1A		G
2	Cation D001A exch sample S2A		G

TABLE 9.3-3 (Cont.)

Sample	Sample Point	Figure	Parameters
3	Anion D002A exch sample S3A		C, G
4	Anion D002A exch sample S4A		C, G
5	Cation D001B exch sample S1B		G
6	Cation D001B exch sample S2B		G
7	Anion D002B exch sample S3B		C, G
8	Anion D002B exch sample S4B		C, G
9	Dilute Caustic Sample S-12		C, G
10	Dilute Sulfuric Acid sample S-13		C, G
r.	Liquid Radwaste System		
SG17-N465	Distillate smpl pmp radn smpl	11.2-4	[illegible], G
SG17-N461A	Chem wste evap concentrate out	11.2-3	G
SG17-N461B	Flr. drn evap concentrate out	11.2-3	G
SG17-N462A	Chem wste evap distillate out	11.2-3	G
SG17-N462B	Flr drn evap distillate out	11.2-3	G

TABLE 9.3-3 (Cont.)

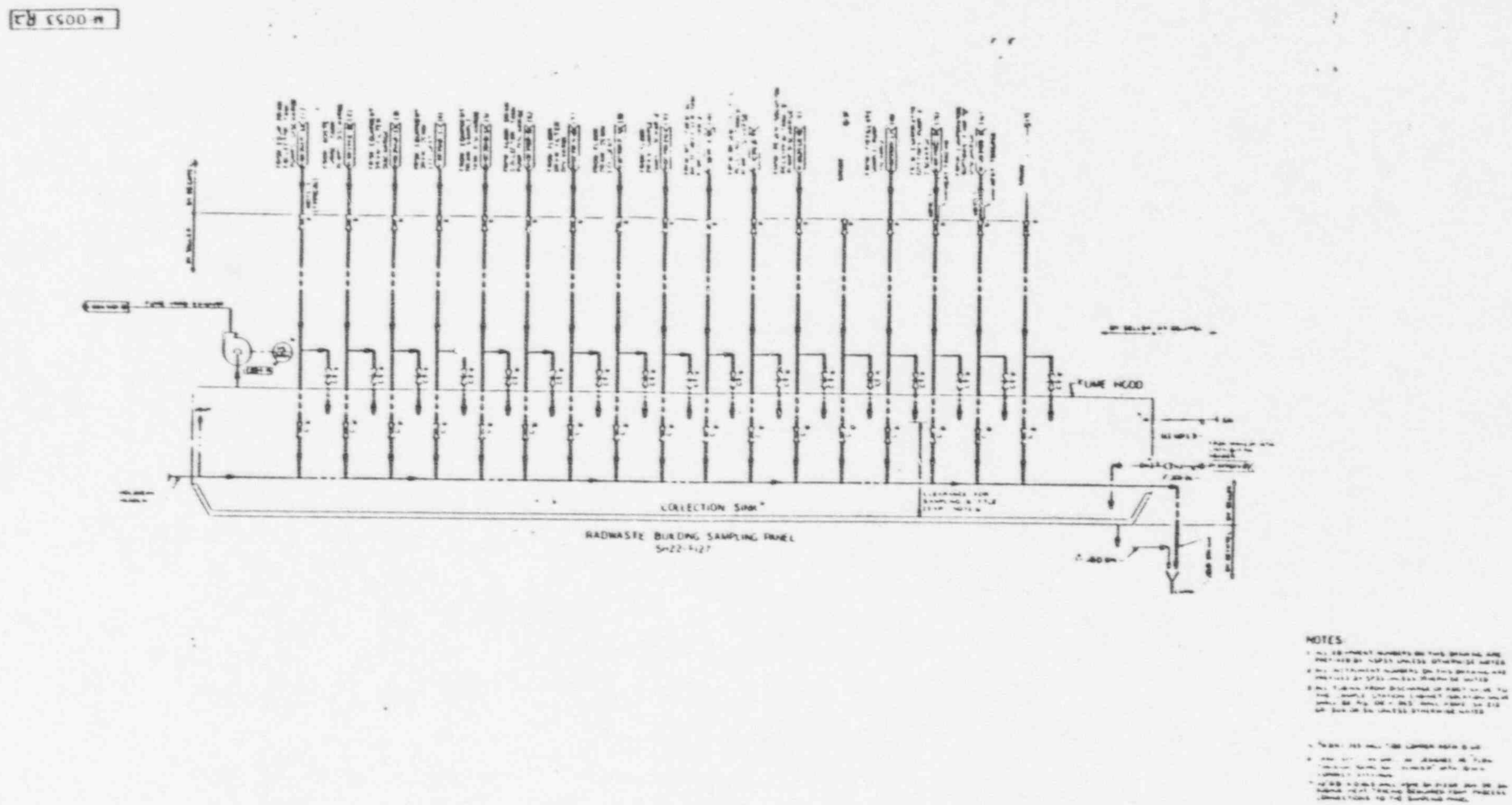
Sample	Sample Point	Figure	Parameters
S.	Post Accident Sample Station at El. 93'-0" (Turbine Bldg)		
	Containment Atmosphere	7.5-5	G, R, H*
	Drywell Atmosphere	7.5-5	G, R, H*
	Recirc. Loop "B"	5.4-2	G, R, H, O_2, C, pH, Cl_2
	Jet Pump Line "D"	5.4-3	G, R, H, O_2, C, pH, Cl_2
	RHR Loop "A"	5.4-17	G, R, H, O_2, C, pH, Cl_2
	RHR Loop "B"	5.4-16	G, R, H, O_2, C, pH, Cl_2
	Suppression Pool	6.2-82	G, R, H, O_2, C, pH, Cl_2

* Hydrogen analyzers are provided by the Combustible Gas Control System

TABLE 9.3-3 (Cont.)

Symbols used under parameters in Table 9.3-3:

C	Conductivity	320.23
Cl_2	Residual chlorine	
G	Grab sample for laboratory analysis for gross radioactivity	320.23
Na	Sodium	
O_2	Dissolved oxygen	
pH	pH	
R	Principle identification and concentration of radionuclides and alpha emitters	320.23
T	Tritium radioactivity (carbon-14 analysis)	
Turb	Turbidity	
S	Integrated sample (suspended solids on in-line membrane filters)	



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROCESS SAMPLING SYSTEM
PART 4

FIGURE 9.3-8A

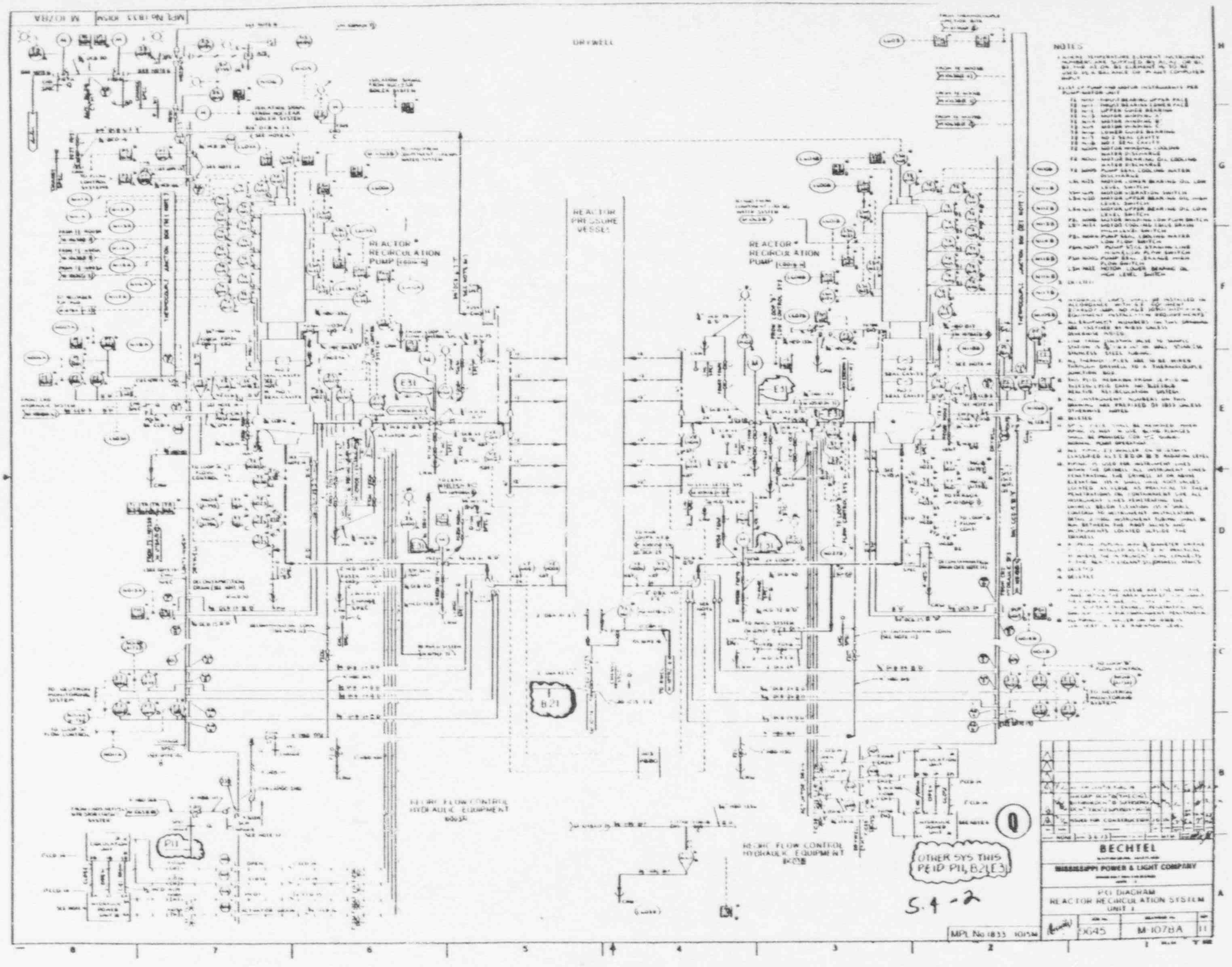
Post Accident Sampling Station

(P&ID to be provided later)

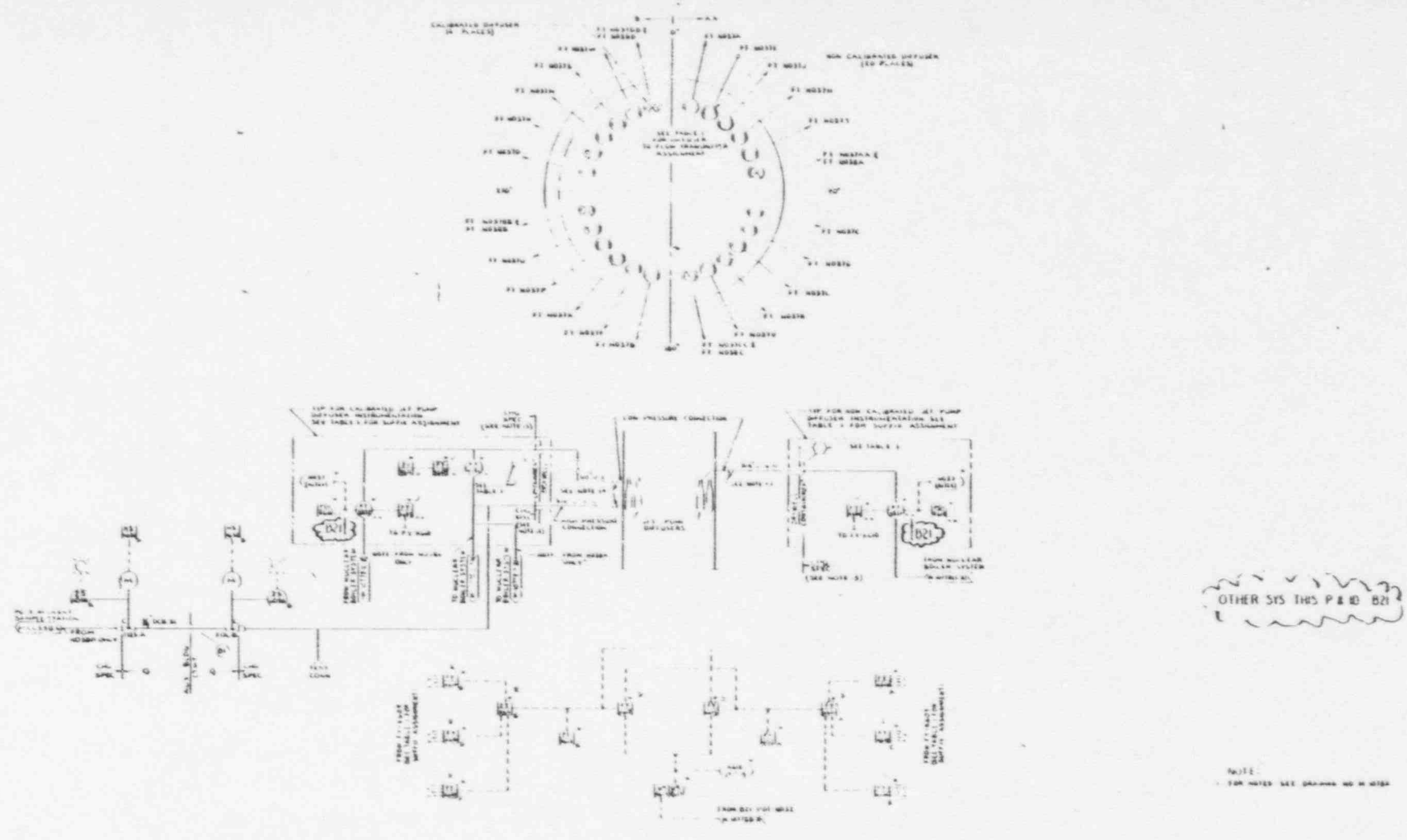
MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROCESS SAMPLING SYSTEM
PART 5

FIGURE 9.3-8B



M-1078A
MPL No 1833 1015M
DRYWELL
REACTOR PRESSURE VESSEL
REACTOR RECIRCULATION PUMP
NOTES
E31
B21
P11
RECIRC FLOW CONTROL HYDRAULIC EQUIPMENT
OTHER SYS THIS P&ID P11, B21, E31
5.4-2
0
BECHTEL
MISSISSIPPI POWER & LIGHT COMPANY
P&I DIAGRAM
REACTOR RECIRCULATION SYSTEM
UNIT 1
9645
M-1078A
11



NOTE:
1. FOR NOTES SEE DRAWING NO M-1076A

TABLE 1 JET PUMP DIFFUSER INSTRUMENTATION SUFFIX LETTER ASSIGNMENT

[illegible]	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24
FT [illegible]	A	E	J	N	T	AA	C	G	L	R	V	CC	B	F	K	P	U	BB	D	H	M	S	W	DD
[illegible]	A	E	J	N	T	AA	C	G	L	R	V	CC	B	F	K	P	U	BB	D	H	M	S	W	DD
[illegible]	A	E	J	N	T	AA	C	G	L	R	V	CC	B	F	K	P	U	BB	D	H	M	S	W	DD
[illegible]	[illegible]																							
COMPUTER INPUT [illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
FT [illegible]	—	—	—	—	—	A	—	—	—	—	—	C	—	—	—	—	—	B	—	—	—	—	—	D
[illegible]	—	—	—	—	—	A	—	—	—	—	—	C	—	—	—	—	—	B	—	—	—	—	—	D
[illegible]	—	—	—	—	—	A	—	—	—	—	—	C	—	—	—	—	—	B	—	—	—	—	—	D
PENETRATION [illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
PENETRATION [illegible]	—	—	—	—	—	[illegible]	—	—	—	—	—	[illegible]	—	—	—	—	—	[illegible]	—	—	—	—	—	[illegible]

5.4-3

Q

BECHTEL

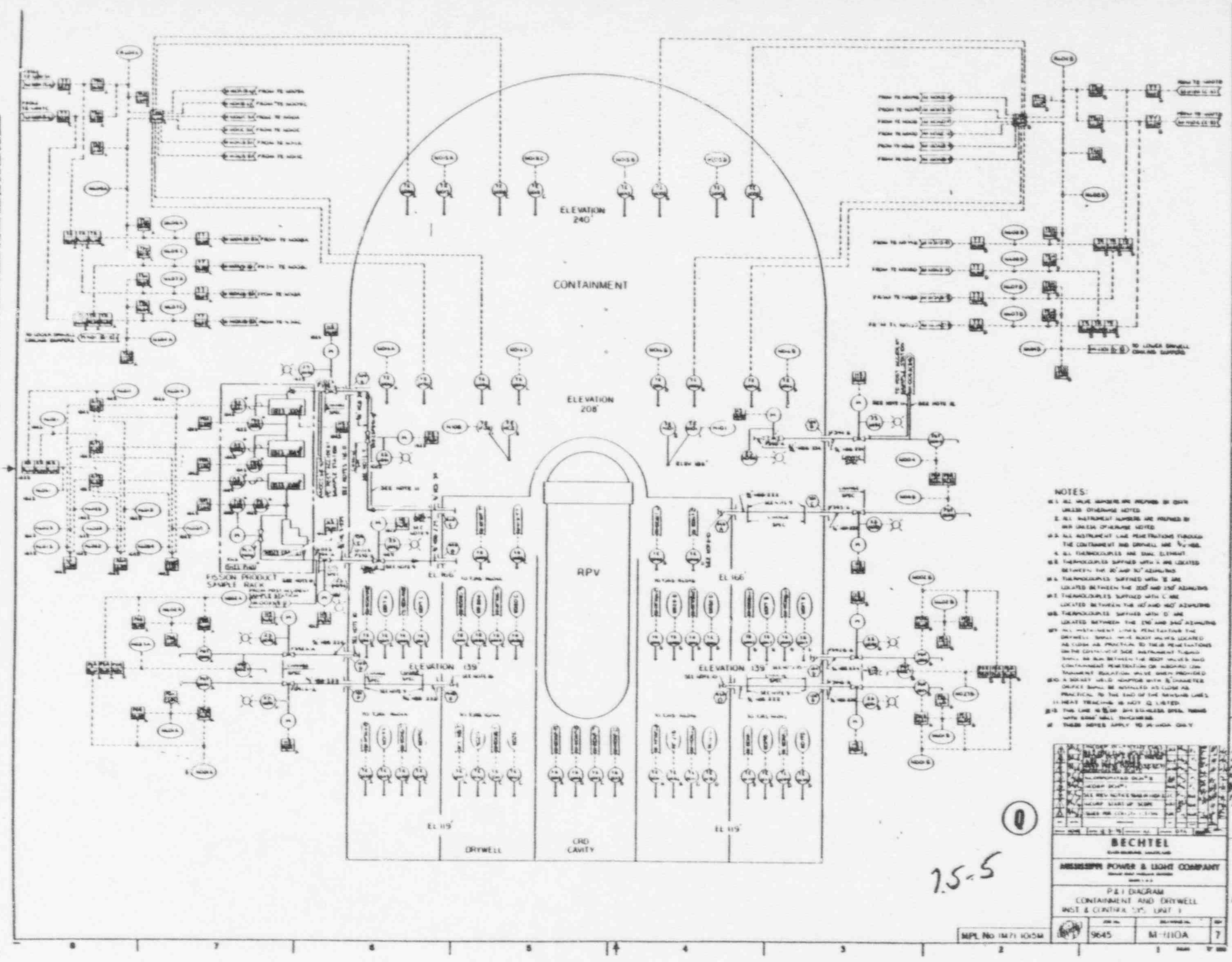
MISSISSIPPI POWER & LIGHT COMPANY

P&I DIAGRAM
REACTOR RECIRCULATION SYSTEM
UNIT 1

MPL No [illegible]

9645

M-1076B



ELEVATION 240'
CONTAINMENT
ELEVATION 208'
RPV
FISSION PRODUCT SAMPLE RACK
EL 166'
ELEVATION 139'
EL 119'
DRYWELL
CRD CAVITY
NOTES:
4. ALL THERMOCOUPLES ARE DUAL ELEMENT
11. HEAT TRACING IS NOT Q LISTED
BECHTEL
MISSISSIPPI POWER & LIGHT COMPANY
P & I DIAGRAM
CONTAINMENT AND DRYWELL
INST & CONTROL SYS UNIT 1
MPL No 1M71 1015M
9645
M-1110A
7
7.5-5

Chap 6

~~Attachment 1~~

shell, 8 x 16 Tyler mesh, having chemical and physical properties which are rated, tested, and qualified in accordance with the requirements of NRC Regulatory Guide 1.52. Refer to subsection 6.5.1 for additional discussion of these items.

In consideration of the possibility of iodine desorption and charcoal ignition at elevated temperatures, a water spray system is provided for the charcoal adsorber section of the SGTS charcoal filter trains. In order to detect any abnormal temperature rise at the outlet of the charcoal adsorber, each charcoal bed is provided with temperature sensors. At the first setpoint there is an alarm in the control room which alerts the operator that an off-design temperature level exists. When the second setpoint is reached, another alarm sounds, the exhaust fan automatically shuts down, and the water deluge valve is automatically activated. The water deluge system is of seismic Category I design.

Insert A attached →

In addition to the instrumentation described above, the filter trains are instrumented to indicate differential pressure across each element of the filter train. Differential pressure is recorded and an alarm provided in the control room for the HEPA filters and the charcoal adsorber. The fan flow rate is recorded in the control room. A low flow switch automatically initiates the standby filter system. Refer to subsection 7.3.1.1.8 for further discussion of the instruments and controls of the SGTS.

During normal plant operation, the standby gas treatment system does not operate except for regularly scheduled testing prescribed by technical specification. Post-operational surveillance is as defined by technical specifications and Regulatory Guide 1.52.

The standby gas treatment system operates in response to the following signals:

a. High pressure in the drywell

b. Low water level in the reactor vessel

c. High radiation in the fuel handling area ventilation system exhaust or the fuel pool sweep system exhaust

Following any of these standby gas treatment system actuation signals, both enclosure building recirculation fans, if available, and both charcoal filter train fans, if available, start. All ventilation penetrations through the auxiliary building walls, including fuel handling area penetrations, are automatically closed on a) or b) above. Closure of the fuel handling area ventilation penetration isolation valves actuates limit switches and automatically shuts down the following equipment:

a. Fuel handling area supply fan

b. Fuel handling area exhaust fan

c. Fuel pool sweep supply fan (if operating)

Insert A

6.5.3.2

A radiation monitoring system is also provided for ~~the~~ SGTS A&B to monitor the effluent radioactivity release ~~discharge~~ to the environment. For further discussion of this system refer to subsection 11.5.2.2.9.

Chng 5

7.3.1.1.8.5 Sequencing

Refer to subsection 8.3.1.1. for a discussion of ESF bus load sequencing. There is no other automatic sequencing in the SGTS.

7.3.1.1.8.6 Redundancy

Two completely independent and redundant SGTS are provided, including independent and redundant logic systems and mechanical equipment. The two logic systems and their associated mechanical devices are powered from separate ESF buses. Physical and electrical separation is maintained between the two systems.

7.3.1.1.8.7 Diversity

Diversity is assured by providing four independent initiation signals for the SGTS. Refer to subsection 7.3.1.1.8.1.

7.3.1.1.8.8 Actuated Devices

Component devices actuated by the SGTS are shown in Table 7.3-23.

7.3.1.1.8.9 Supporting Systems

The supporting systems required for SGTS operation are identified in Table 7.3-1 and are described as referenced in that table.

7.3.1.1.8.10 Nonessential Components

Computer and Annunciator Components

> The SGTS logic and control systems are electrically isolated from the plant computer and annunciator systems. Failure of the computer or annunciator will have no effect on the operation of the SGTS.

See Insert 1 → attached

7.3.1.1.9 Suppression Pool Makeup System

Refer to subsection 6.2.7 for a description of the suppression pool makeup system. The suppression pool makeup system provides water from the upper containment pool to the suppression pool by gravity flow after a LOCA.

Final system drawings for the SPMU system are provided by reference in Section 1.7. The logic diagram is Drawing J-1279. The electrical schematic is Drawing E-1220. The system P&ID is provided as Figure 6.2-82.

7.3.1.1.9.1 Initiating Circuits

The suppression pool makeup system is automatically initiated 30 minutes after a LOCA is detected or on low-low suppression pool level following a LOCA. It can also be manually initiated provided a LOCA has occurred.

Insert 1

7.3.1.1.8.10

The SGTS A&B ~~exhaust~~ vents are also provided with an ~~radiation~~ effluent radioactivity monitoring system that alarms in the control room upon reading a high radiation level. See subsection 11.5.2.2.9 for a further description of this system.

b. Ventilation Systems - radiation monitoring subsystems

1. Containment/drywell ventilation exhaust radiation monitoring subsystem

2. Auxiliary building - fuel handling area ventilation exhaust - radiation monitoring subsystem

3. Auxiliary building fuel handling area pool sweep exhaust - radiation monitoring subsystem

4. Control room intake - radiation monitoring subsystem

Area and airborne radiation monitors are discussed in section 12.3.4. The following non-safety-related process radiation monitors are discussed in Section 11.5:

a. For gaseous effluent streams

1. Containment ventilation exhaust RMS

2. Radwaste building ventilation RMS

3. Fuel handing area ventilation RMS

4. Turbine building ventilation RMS

5. Standby Gas Treatment Exhaust ventilation RMS

b. For liquid effluent streams

1. Radwaste effluent RMS A/B

c. For gaseous process streams

1. Offgas pretreatment RMS

2. Offgas post-treatment RMS

3. Carbon bed vault RMS

d. For liquid process streams

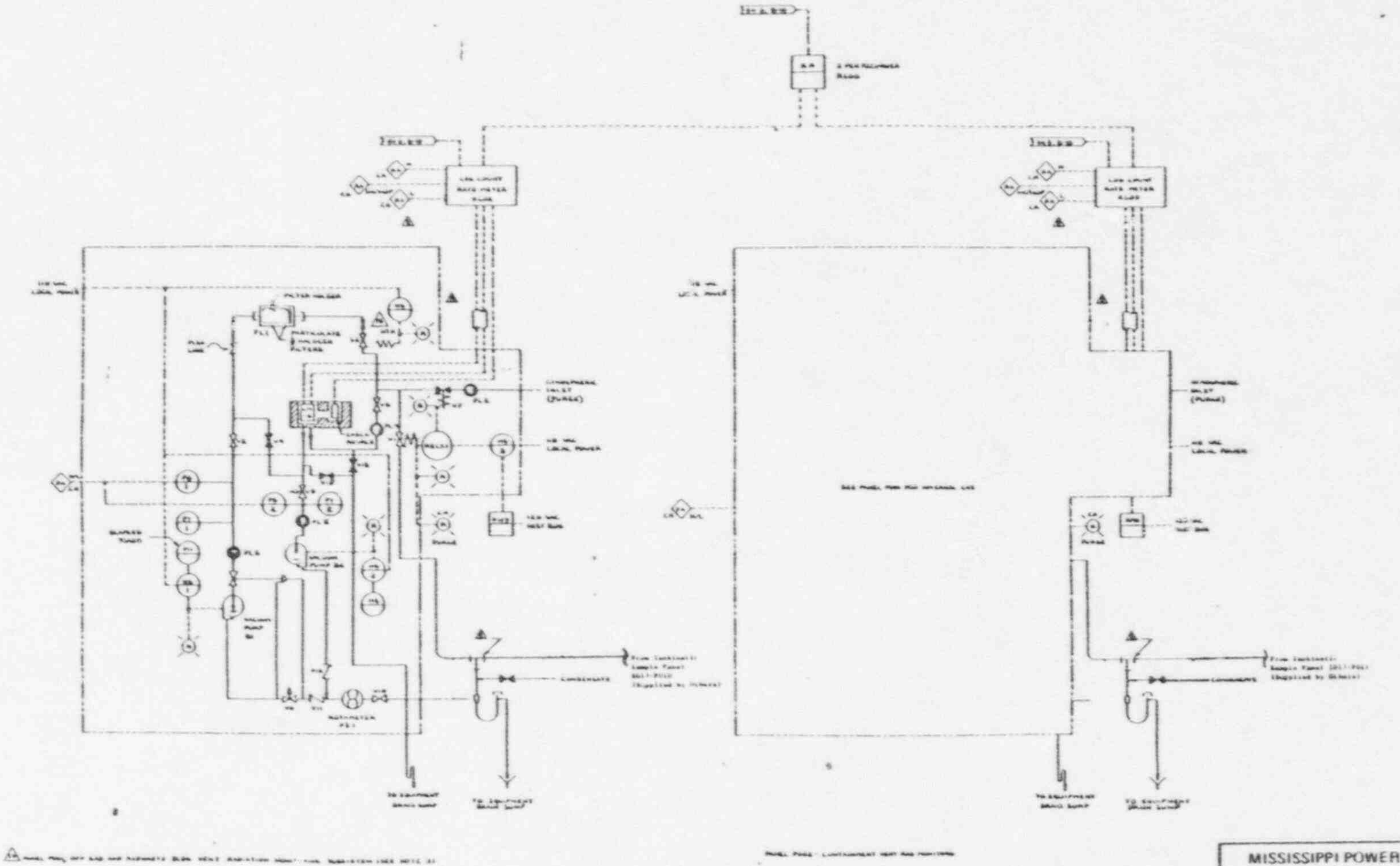
1. Service water system RMS (loops A and B)

2. Component cooling water RMS

The process radiation monitoring system is shown in Figure 7.6-1.

7.6.1.2.1 Main Steam Line Radiation Monitoring Subsystem

The main steam line radiation subsystem is discussed in subsections 7.2.1, 7.3.1, and 11.5.1.1.1.

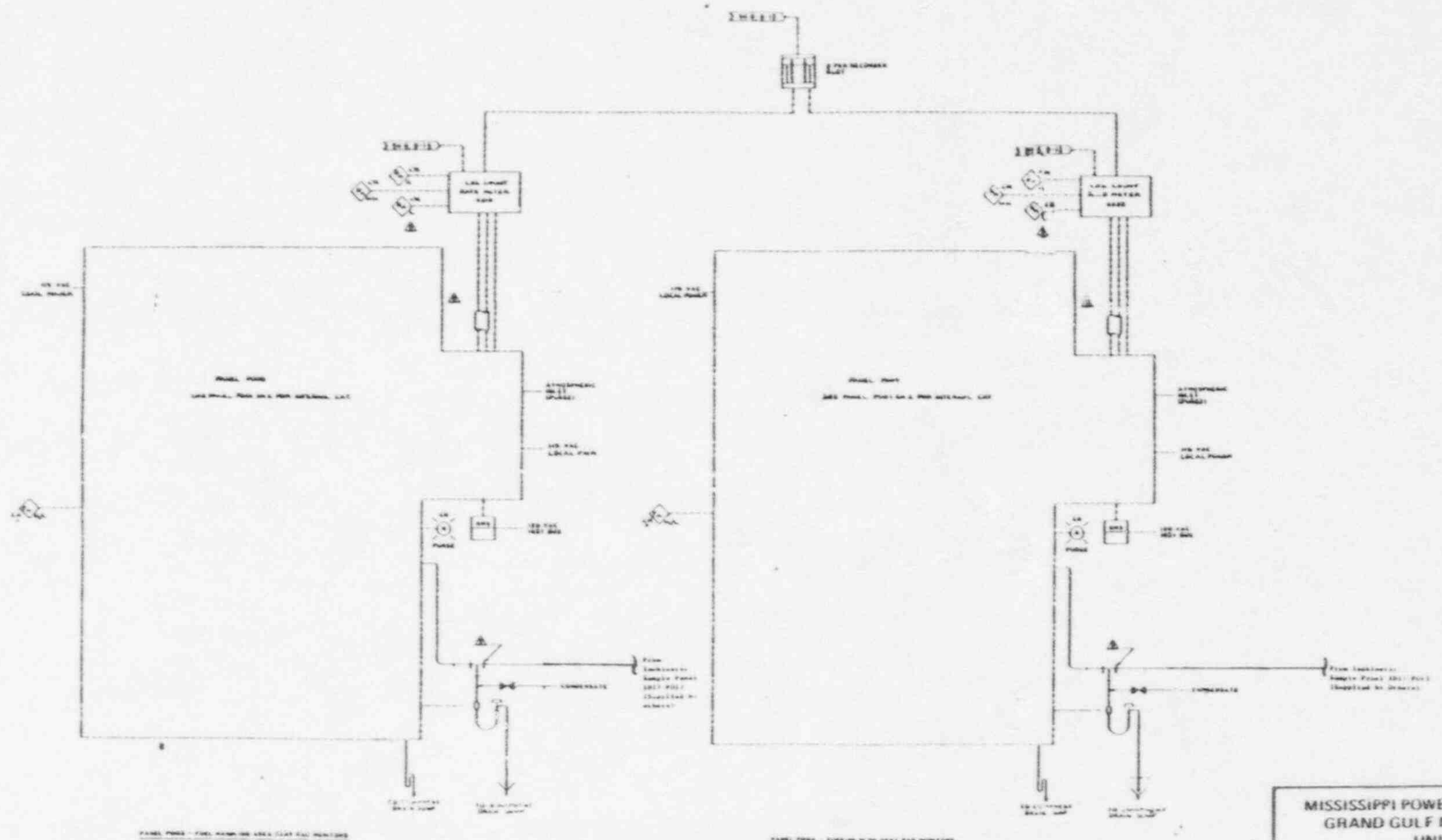


MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROCESS RADIATION MONITORING SYSTEM

FIGURE 7.6-1b

Amend. 27 1/79



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

PROCESS RADIATION MONITORING SYSTEM

FIGURE 7.6-1c

Amend. 27 1/79

Chap 8?

A high-high-radiation signal actuates an alarm, automatically initiates isolation of the auxiliary building and the fuel handling area ventilation systems, and starts the standby gas treatment system. A high signal only alarms and a low-radiation signal indicates failure.

Insert A → Attached

The auxiliary building pressure control operates to maintain the fuel pool area in the auxiliary building at the required pressure and provides the operator with sufficient indication to maintain proper operating conditions. (See subsection 7.7.1.9 for a description of auxiliary building pressure control.)

Differential pressure switches are provided across the air supply and exhaust fans to start the standby fan on low differential pressure. Supply and exhaust fans are interlocked to ensure that a supply fan does not start unless an exhaust fan is running.

Temperature controls are provided to maintain the incoming air at the desired temperature.

The supply and exhaust fan discharge and suction dampers automatically open when the fan is started and close when the fan is stopped.

Differential pressure indicators are provided across filters for maintenance purposes.

Fire protection is described in subsection 9.5.1.

9.4.3 Radwaste Building Ventilation System

9.4.3.1 Design Bases

9.4.3.1.1 Power Generation Design Bases

The radwaste building ventilation system is a nonsafety-related system designed to provide an environment with controlled temperature and air-flow patterns to ensure both the comfort and safety of plant personnel and the integrity of equipment and components.

The system design is based on outdoor summer conditions of 95 F dry-bulb and 79 F wet-bulb temperatures. Summer indoor design temperatures include 72 F in the radwaste control station, a maximum temperature of 104 F in areas that may be occupied, and 120 F in the equipment cells. Winter indoor design temperatures include 65 F in occupied areas, 72 F in the radwaste control station, and 50 F in the equipment cells, based on an outdoor design temperature of 15 F.

Once-through ventilation is employed for space cooling and air-flow control from areas of low potential radioactivity to higher activity areas. Outside air is filtered, tempered, and delivered to the clean areas such as the radwaste control station and

Insert "A"

9.4.2.5

A separate radiation monitoring system is provided to monitor effluent radioactivity releases to the environment. For a further discussion of this system see sub section 11.5.2.2.7

The exhaust air to the radwaste building vent is provided with a radiation monitoring system that alarms in the control room and automatically shuts down the ventilation system upon reaching a high-radiation level.

See Insert A Attached

9.4.4 Turbine Building Ventilation System

9.4.4.1 Design Bases

9.4.4.1.1 Power Generation Design Bases

The turbine building ventilation system consists of heating, ventilation, and cooling systems designed to provide an environment with controlled temperature and humidity to ensure both the safety of plant personnel and the integrity of equipment and components.

System design is based on outdoor summer conditions of 95 F dry bulb temperature and 79 F wet bulb temperature. Indoor design temperatures are 100 F for the area above the operating floor; 96 F for the areas below the operating floor, and 104 F for the elevator machine room.

It is a design objective to limit the relative humidity in the turbine building at 50 percent. An indoor minimum temperature of 65 F is maintained during winter shutdown conditions, based on an outside air design temperature of 15 F. Electric heating coils are sized based on an outside air design temperature of 20 F.

Air-flow control from areas of low potential radioactivity to areas of high potential radioactivity is based on infiltrating outside air above the turbine building operating floor, allowing it to flow down through the lower decks and condenser area into a filter train and, finally, exhausting it through the turbine building vent. Offsite dose calculations of subsection 11.3 indicate filtration is not required, therefore, filter train internals are not installed and operated for credit.

9.4.4.2 System Description

The turbine building ventilation system is shown in Figures 9.4-6 and 9.4-7. The major system components and associated fabrication and performance data are given in Table 9.4.6. Filter train components are not installed and operated for credit and are shown for information only. If at a later date the need for filtration in indicated, the filters may be operated.

The outside air is introduced into the turbine building above the operating floor and then drawn down to the areas below the floor by the exhaust system. The turbine building is maintained at a negative pressure with respect to the outside to assure that no outleakage of air will occur.

Insert "A"

9.4.3.5 For a further description of the radwaste building ~~bldg~~ ventilation ~~radiation~~ radioactivity monitoring system see section 11.5.2.2.6.

The system has been designed in accordance with the codes and standards given in Table 3.2-1, Item XXXVII.

The isolation valve closure time for all eight containment isolatic valves associated with containment ventilation and filtration is 4 seconds after receiving the signal to close. 021.02

A radiation monitoring system is provided to detect high radiation in the containment ventilation exhaust and drywell purge exhaust. A high-radiation signal actuates an alarm and automatically initiates isolation of the containment and drywell.

Insert A → attached

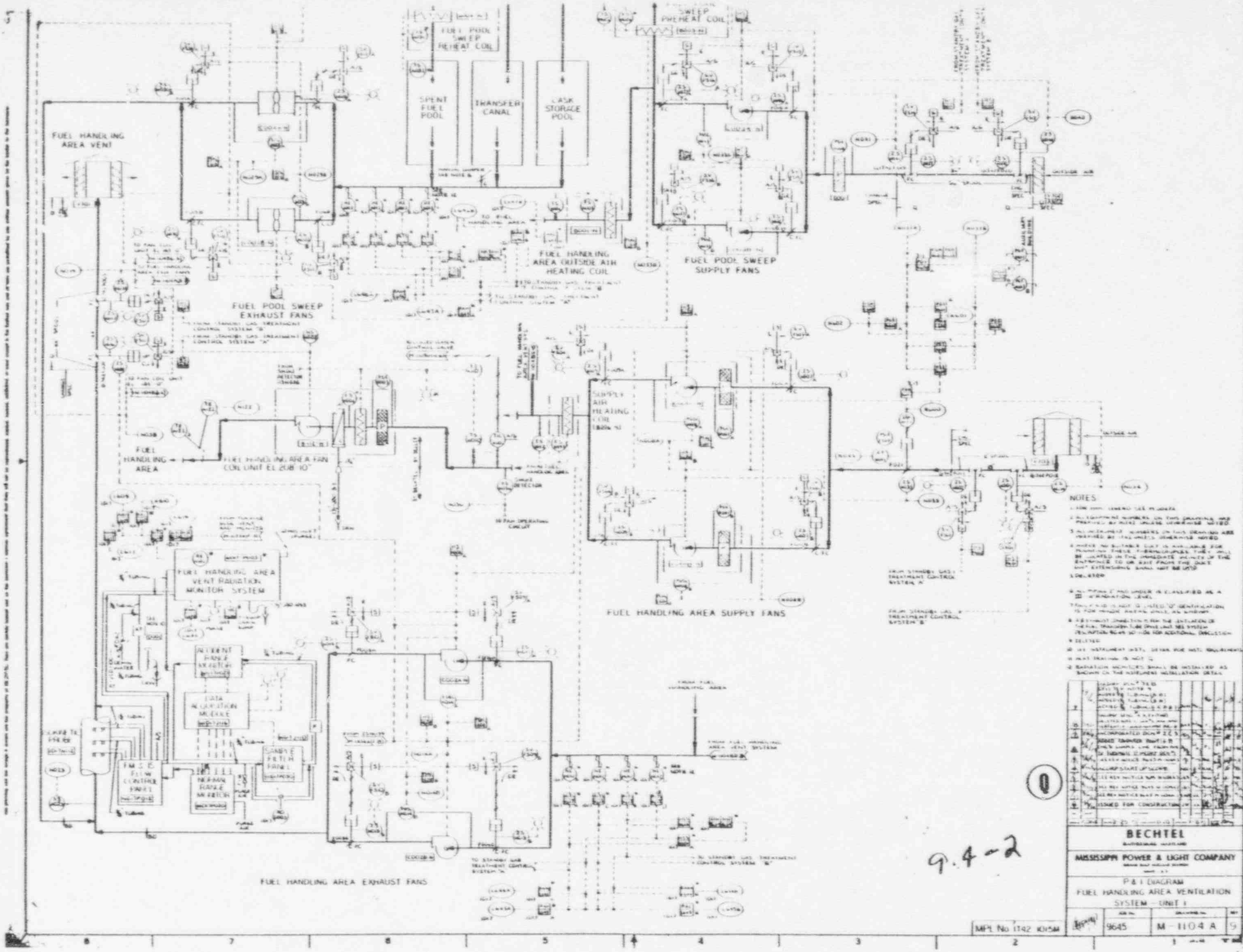
9.4.7.4 Tests and Inspections

The cooling and ventilating systems are periodically inspected to assure that all normally operating equipment is functioning

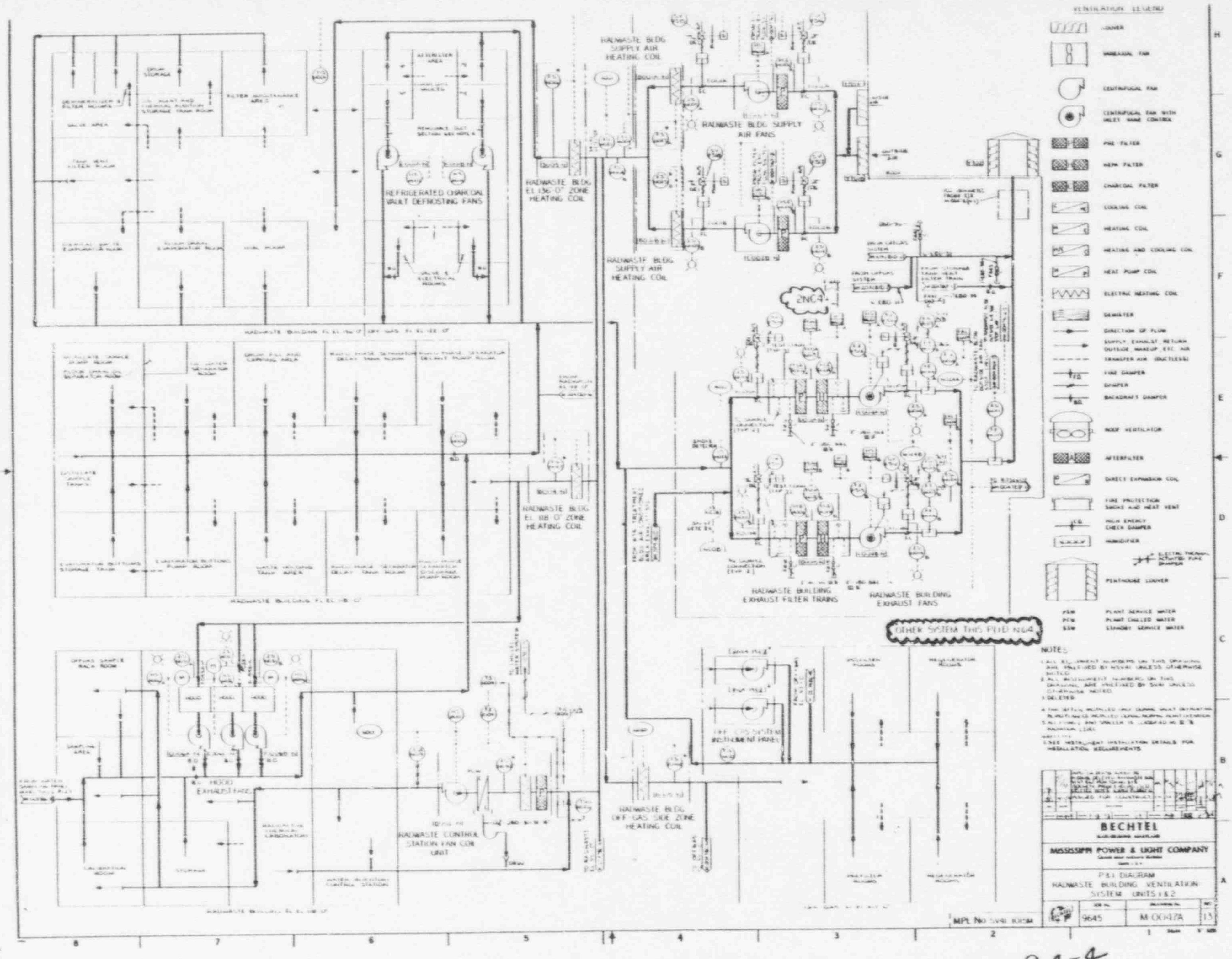
Insert "A"

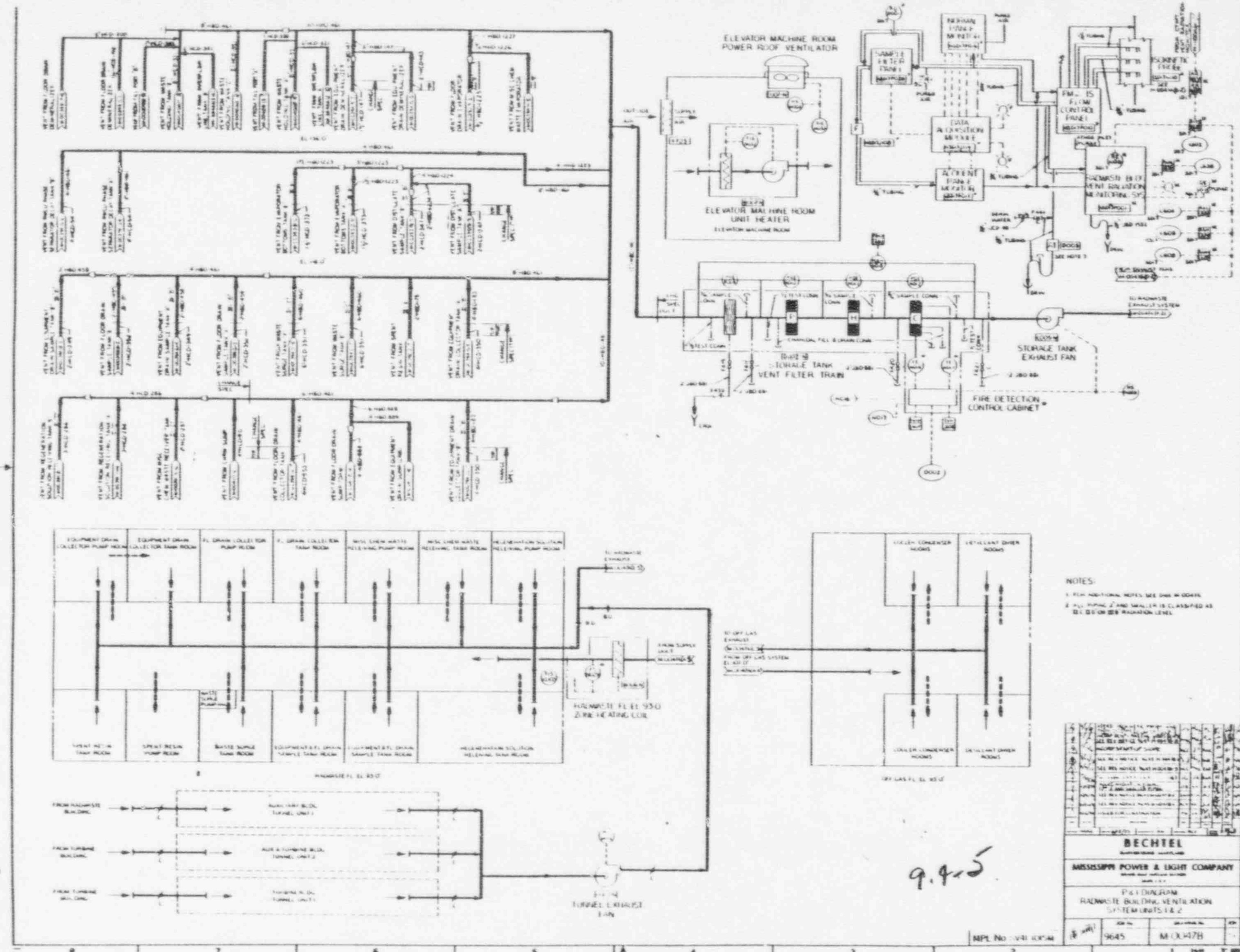
9.4.7.3

A separate radiation monitoring system is also provided to monitor effluent ~~radiation~~ radioactivity releases to the environment. For a further discussion of this system see subsection 11.5.2.2.4.

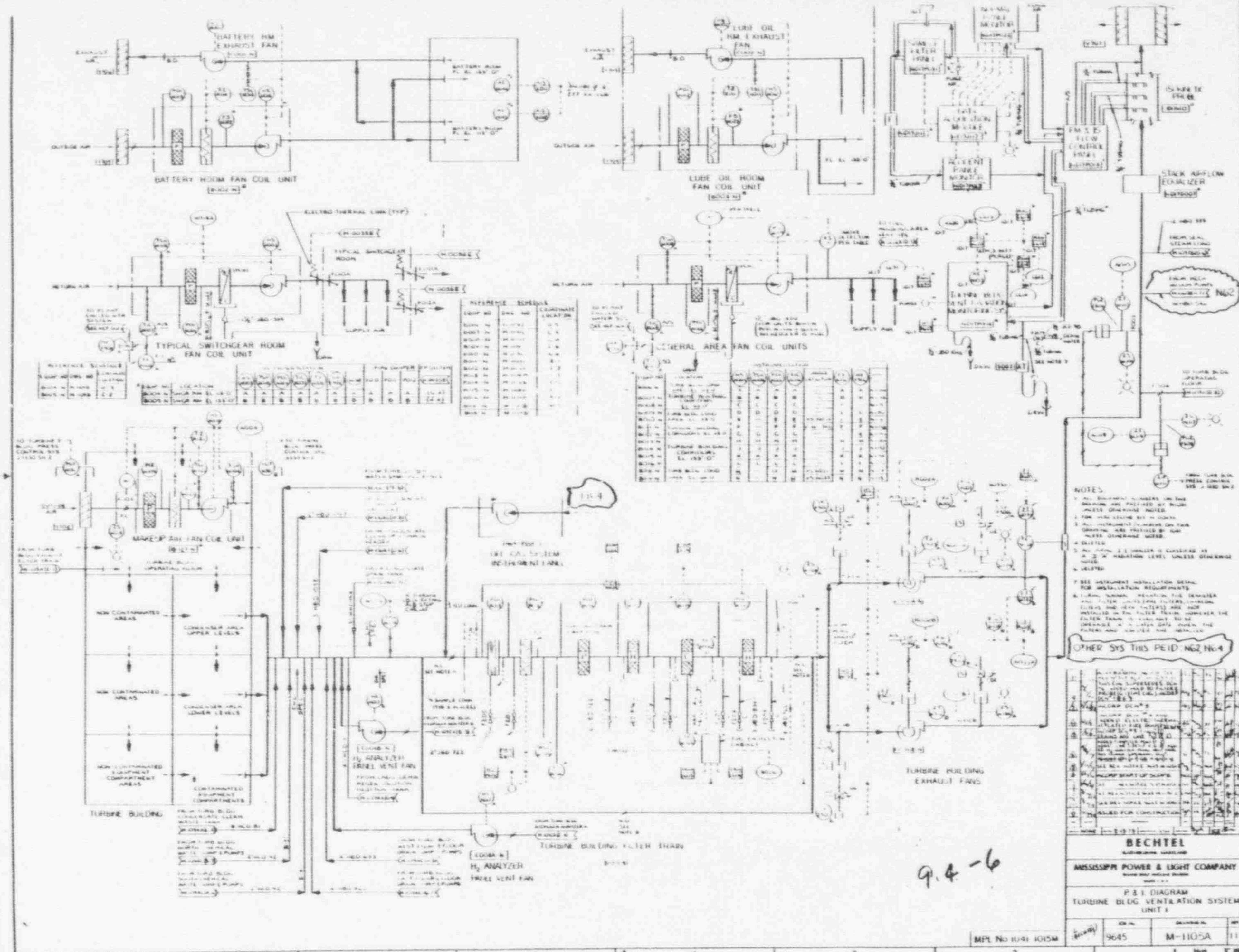


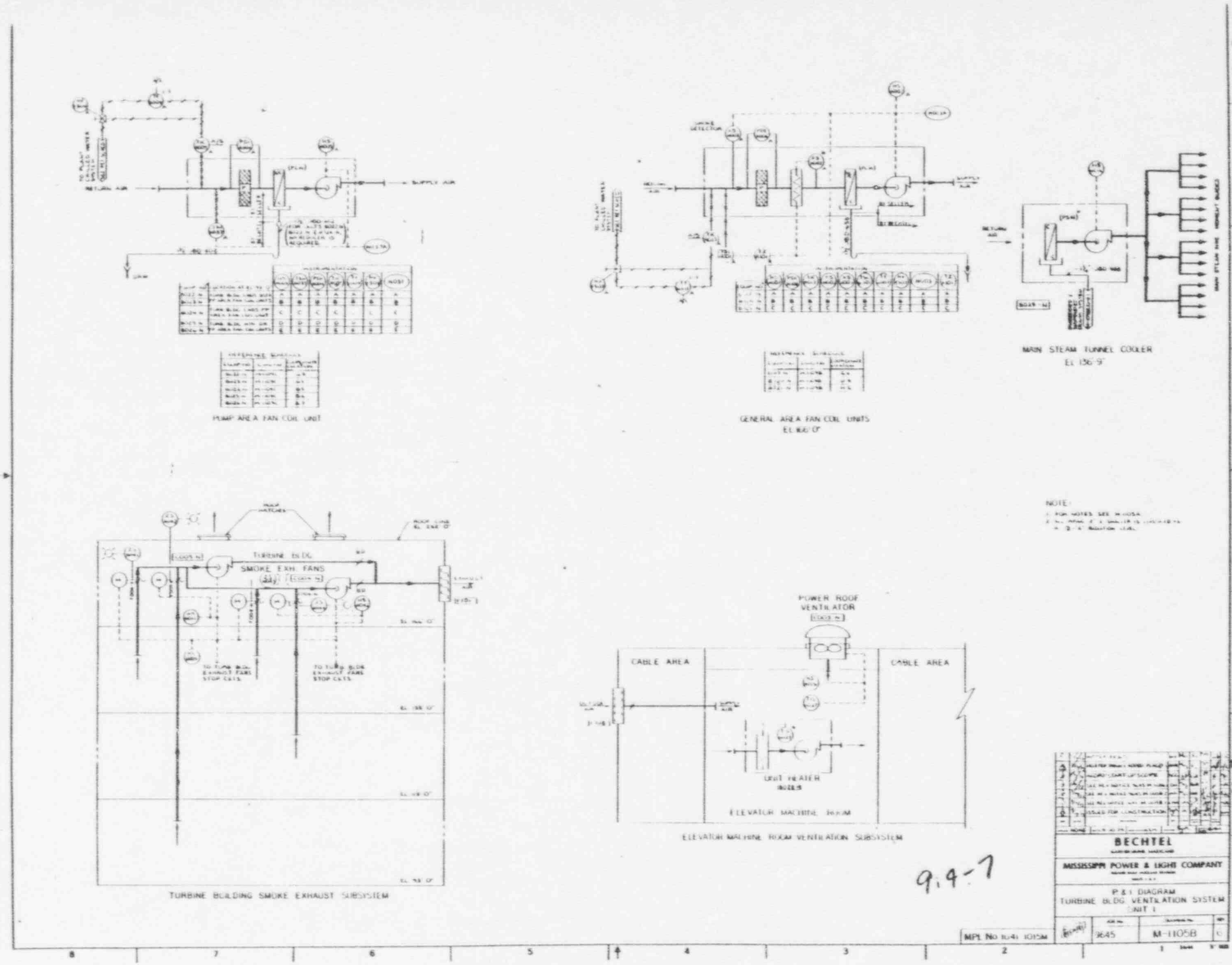
FUEL HANDLING AREA VENT
FUEL POOL SWEEP REHEAT COIL
SPENT FUEL POOL
TRANSFER CANAL
CASK STORAGE POOL
SWEEP PREHEAT COIL
FUEL POOL SWEEP EXHAUST FANS
FUEL HANDLING AREA OUTSIDE AIR HEATING COIL
FUEL POOL SWEEP SUPPLY FANS
FUEL HANDLING AREA
FUEL HANDLING AREA FAN COIL UNIT EL 208'10"
SUPPLY AIR HEATING COIL
FUEL HANDLING AREA SUPPLY FANS
FUEL HANDLING AREA VENT RADIATION MONITOR SYSTEM
ACCIDENT RANGE MONITOR
DATA ACQUISITION MODULE
SAMPLE FILTER PANEL
NORMAL RANGE MONITOR
FLOW CONTROL PANEL
FUEL HANDLING AREA EXHAUST FANS
NOTES
ISSUED FOR CONSTRUCTION
BECHTEL
MISSISSIPPI POWER & LIGHT COMPANY
P & I DIAGRAM
FUEL HANDLING AREA VENTILATION
SYSTEM — UNIT 1
9645
M-1104 A
9
MPL No 1T42 IO15M
9.4-2


VENTILATION LEGEND
RADWASTE BLDG SUPPLY AIR HEATING COIL
RADWASTE BLDG SUPPLY AIR FANS
REFRIGERATED CHARCOAL VAULT DEFROSTING FANS
RADWASTE BLDG EL 136'-0" ZONE HEATING COIL
RADWASTE BLDG SUPPLY AIR HEATING COIL
2NC4
RADWASTE BLDG EL 118'-0" ZONE HEATING COIL
RADWASTE BUILDING EXHAUST FILTER TRAINS
RADWASTE BUILDING EXHAUST FANS
OTHER SYSTEM THIS P&ID N64
HOOD EXHAUST FANS
RADWASTE CONTROL STATION FAN COIL UNIT
OFF-GAS SYSTEM INSTRUMENT PANEL
RADWASTE BLDG OFF-GAS SIDE ZONE HEATING COIL
NOTES
BECHTEL
MISSISSIPPI POWER & LIGHT COMPANY
P&I DIAGRAM
RADWASTE BUILDING VENTILATION SYSTEM UNITS 1 & 2
9645
M-0047A
13
MPL No. 5V41 1010M
9.4-4

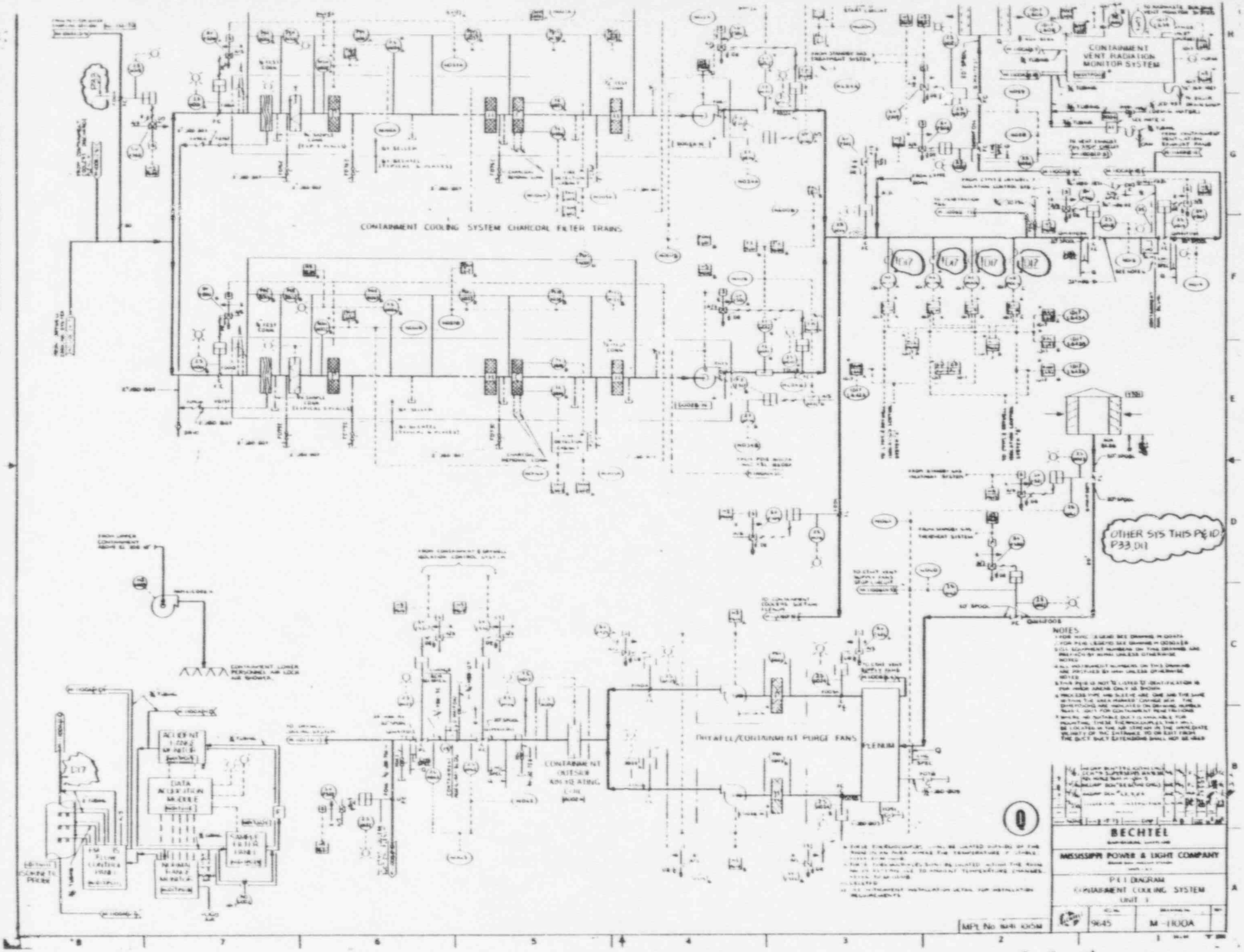


ELEVATOR MACHINE ROOM
POWER ROOF VENTILATOR
ELEVATOR MACHINE ROOM
UNIT HEATER
SAMPLE FILTER PANEL
DATA ACQUISITION MODULE
ACCIDENT RANGE MONITOR
FM-15 FLOW CONTROL PANEL
ISOKINETIC PROBE
RADWASTE BLDG VENT RADIATION MONITORING SYS
STORAGE TANK
VENT FILTER TRAIN
STORAGE TANK
EXHAUST FAN
FIRE DETECTION
CONTROL CABINET
RADWASTE FL EL 93.0
ZONE HEATING COIL
TUNNEL EXHAUST FAN
NOTES:
P & I DIAGRAM
RADWASTE BUILDING VENTILATION
SYSTEM UNITS 1 & 2
BECHTEL
MISSISSIPPI POWER & LIGHT COMPANY
9645
M-0047B
9.4-5


BATTERY RM EXHAUST FAN
BATTERY ROOM FAN COIL UNIT
LUBE OIL RM EXHAUST FAN
LUBE OIL ROOM FAN COIL UNIT
ISOKINETIC PROBE
FMS 15 FLOW CONTROL PANEL
STACK AIRFLOW EQUALIZER
ACCIDENT RANGE MONITOR
TYPICAL SWITCHGEAR ROOM FAN COIL UNIT
ELECTRO-THERMAL LINK (TYP)
GENERAL AREA FAN COIL UNITS
REFERENCE SCHEDULE
TURBINE BLDG VENT RADIATION MONITORING SYS
MAKEUP AIR FAN COIL UNIT
OFF GAS SYSTEM INSTRUMENT PANEL
TURBINE BUILDING FILTER TRAIN
TURBINE BUILDING EXHAUST FANS
H2 ANALYZER PANEL VENT FAN
TURBINE BUILDING
NON CONTAMINATED AREAS
CONDENSER AREA UPPER LEVELS
CONDENSER AREA LOWER LEVELS
CONTAMINATED EQUIPMENT COMPARTMENTS
NOTES:
OTHER SYS THIS P&ID: N62 N64
BECHTEL
MISSISSIPPI POWER & LIGHT COMPANY
P & I DIAGRAM
TURBINE BLDG VENTILATION SYSTEM
UNIT 1
9645
M-1105A
11
MPL No 1041 1015M
9.4-6



RETURN AIR
SUPPLY AIR
PUMP AREA FAN COIL UNIT
GENERAL AREA FAN COIL UNITS
EL 166'0"
MAIN STEAM TUNNEL COOLER
EL 136'9"
NOTE:
TURBINE BLDG
SMOKE EXH. FANS
TO TURB. BLDG. EXHAUST FANS STOP CKTS
TURBINE BUILDING SMOKE EXHAUST SUBSYSTEM
POWER ROOF VENTILATOR
CABLE AREA
CABLE AREA
UNIT HEATER
ELEVATOR MACHINE ROOM
ELEVATOR MACHINE ROOM VENTILATION SUBSYSTEM
9.4-7
BECHTEL
MISSISSIPPI POWER & LIGHT COMPANY
P & I DIAGRAM
TURBINE BLDG VENTILATION SYSTEM
UNIT 1
9645
M-1105B
MPL No 1641 1015M


CONTAINMENT COOLING SYSTEM CHARCOAL FILTER TRAINS
CONTAINMENT VENT RADIATION MONITOR SYSTEM
OTHER SYS THIS P&ID P33,D11
DRYWELL/CONTAINMENT PURGE FANS
PLENUM
CONTAINMENT OUTSIDE AIR HEATING COIL
CONTAINMENT LOWER PERSONNEL AIR LOCK AIR SHOWER
DATA ACQUISITION MODULE
NOTES
BECHTEL
MISSISSIPPI POWER & LIGHT COMPANY
P & I DIAGRAM
CONTAINMENT COOLING SYSTEM
UNIT 1
9645
M-1100A
MPL No. M41-1051M

9.4-11

Chap 11

~~Adjustments~~
to initiate discharge valve isolation on the offgas or liquid radwaste systems if predetermined release rates are exceeded and to provide for sampling at certain radiation monitor locations to allow determination of specific radionuclide content.

The radiation monitoring systems provided to meet these objectives are:

a. For gaseous effluent streams

   1. Containment Ventilation RMS

   2. Offgas and Radwaste Building Ventilation RMS

   3. Fuel Handling Area Ventilation RMS

   4. Turbine Building Ventilation RMS

   5. Standby Gas Treatment/Exhaust Ventilation RMS

b. For liquid effluent streams

   1. Radwaste Effluent RMS

c. For gaseous process streams

   1. Offgas Pretreatment RMS

   2. Offgas Post-treatment RMS

   3. Carbon Bed Vault RMS

d. For liquid process streams

   1. Standby Service Water System RMS (Loops A and B)

   2. Component Cooling Water RMS

11.5.1.2 Design Criteria

11.5.1.2.1 Systems Required for Safety

The design criteria for the safety-related radioactivity monitoring systems are that the systems:

a. Withstand the effect of natural phenomena (e.g., earthquakes) without loss of capability to perform their functions.

b. Perform their intended safety function in the environment resulting from normal and postulated accident conditions.

11.5.2.2.4 Containment Ventilation Radiation Monitoring System

See new writeup

This system monitors the containment ventilation discharge for gross radiation level and collects halogen and particulate samples. A representative sample is continuously extracted from the ventilation ducting through an isokinetic probe in accordance with ANSI N13.1-1969, passed through the containment ventilation sample panel for monitoring and sampling, and returned to the ventilation ducting. The sample panel has a pair of filters (one for particulate collection and one for halogen collection) in parallel (with respect to flow) with a continuous gross radiation detection assembly. The gross radiation detection assembly consists of a shielded chamber, a beta-sensitive GM tube, and a check source. A radiation monitor in the control room analyzes and visually displays the measured gross radiation level.

The sample panel shielded chambers can be purged with room air to check detector response to background radiation by using a three-way solenoid valve operated from the control room. The sample panel measures and indicates sample line flow. A solenoid operated check source operated from the control room can be used to check operability of the gross radiation channel.

Power is supplied from 125 V dc bus D for the radiation monitor and recorder, and from a 120 V ac local bus for the sample panel. The recorder has two pens, one used by this system and the other used by the offgas and radwaste building ventilation radiation monitoring system.

The radiation monitor has three trip circuits: two upscale (high-high and high) and one downscale (low). Each trip is visually displayed on the radiation monitor. These three trips actuate corresponding control room annunciators: containment ventilation high-high radiation, containment ventilation high radiation, and containment ventilation downscale. High or low sample flow measured at the sample panel actuates a control room containment ventilation sample high-low flow annunciator.

11.5.2.2.5 Liquid Process and Effluent Monitoring Systems

These systems monitor the gamma radiation levels of liquid process and effluent streams. With the exception of the radwaste system effluent, the streams monitored normally contain only background levels of radioactive materials. Increases in radiation level may be indicative of heat exchanger leakage or equipment malfunction.

Power is supplied from 125 V dc non-divisional buses for the radiation monitors and recorders, and from a ~~120~~ 110 V ac local bus for the sample panels.

Replace existing section 11.5.2.2.4 with the following

The containment ventilation ~~radiation~~ radioactivity monitoring systems consists of a microprocessor based system and a GE constant flow system both which utilize a single flow monitoring and isokinetic sampling unit located in the exhaust duct. They systems are designed to be operated independently or in parallel.

11.5.2.2.4 Containment Ventilation ~~Radiation~~ Radioactivity Monitoring System

11.5.2.2.4.1 Containment Vent. RADN MONITORING SYSTEM (MICROPROCESSOR BASED)

This system monitors the containment ventilation discharge for ~~gross radiation level~~ gases, iodines, and particulates, and collects halogen and particulate samples. A representative sample is continuously extracted from the ventilation ducting through ~~an isokinetic probe~~ a flow monitoring and isokinetic sample unit in accordance with ANSI N13.1-1969, passed through the containment ventilation sample panel for monitoring and sampling, and returned to the ventilation ducting.

The ~~effluent~~ ~~radiation~~ radioactivity monitoring system consists of ~~an~~ a flow monitoring and isokinetic sample ~~probe~~ unit (FMSIS) located in the exhaust duct, an isokinetic sample panel, a microprocessor based normal range ~~radiation~~ radioactivity monitor, a particulate-iodine sample filter, a microprocessor based accident range ~~radiation~~ radioactivity monitor, a data aquisition module (DAM), and the central control terminal with a report generating computer interface located in the control room. In addition, a second completely redundant control terminal is provided for use in the Technical Support Center.

During normal plant operation, the ~~effluent~~ sample is continuously delivered to the microprocessor based normal range ~~radiation~~ radioac- monitor for particulate, iodine, and noble gas analysis. Should the ~~radiation~~ radioactivity level exceed the normal range monitor's capacity (i. post-accident) the sample flow will

be automatically isolated from the normal range monitor and directed to the sample filter and accident range monitor. The operating ranges of the monitors ~~shall~~ overlap sufficiently to permit continuity of measurement upon changing from the normal to the accident range monitor. Should the ~~radiation~~ radioactivity level return to normal (i.e. ~~post-post~~ pre-accident), the normal range monitor will automatically resume the ~~radiation~~ radioactivity monitoring function.

In addition, the isokinetic sample panel ~~will also~~ is capable of delivering a sample simultaneously to the G.E. constant volume ~~radiation~~ radioactivity monitoring system, if desired. This extends the overall system capabilities since the effluent can still be monitored while the microprocessor based system is out of service. See subsection 11.5.2.2.4.2 for a discussion of the G.E. system.

The FM&IS unit consists of a ~~A~~ velocity sensing (flow monitoring) section consisting of an array of total and static pressure sensors symetrically connected to an averaging manifold to provide for the instantaneous and continuous monitoring of the stack flow rate. ~~[illegible]~~. A minimum of one velocity sensor ~~shall be~~ is provided for each half square foot of duct cross-sectional area. This sensor ~~shall be~~ is capable of measuring the average duct velocity within ± 3 percent of actual flow. The FM&IS unit also consists of a

~~[illegible]~~ ~~A~~ multi-probe isokinetic sampling section consisting of an array of sampling nozzles connected to a collection manifold for extracting a highly representative sample of the stack air from the air stream. The sampling nozzles ~~shall be~~ are capable of simultaneously extracting an equal volume of stack gas and ~~shall be~~ are located such that a minimum of one nozzle exists for each square foot of duct cross-sectional area. The overall accuracy of the isokinetic sampling rate ~~shall be~~ is within ± 5 percent of actual stack flow conditions.

The sample lines from the FM&IS unit to the radioactivity monitoring panels are provided with heat tracing to prevent any entrained water in the air stream from reaching the sampling medium. The heat tracing is shown on Figure 9.4-11.

The ~~radiation~~ radioactivity detection assembly for the normal range monitor consists of a shielded chamber, a sample filter of activated charcoal for iodine collection, a single channel analyzer capable of monitoring the iodine 364 Kev gamma peak with approximately 4% (4π) efficiency, a sample filter of 0.009 inch thick filter paper for particulate collection, a single channel analyzer to monitor the particulate Cs-137 beta particles with approximately 11% (4π) efficiency, a beta scintillation detector and a GM tube to monitor gross radioactivity (ie. noble gas activity). with an accuracy of approximately 15% of logarithmic scale down to 40 Kev, and a check source mechanism.

The normal range monitor is also provided with a purge assembly which can be manually initiated from the data aquisition module or either of the central control terminals. In addition, upon receipt of a high ~~radiation~~ radioactivity isolation signal, the normal range monitor will automatically isolate and the purge will automatically be initiated to ~~flush~~ purge the normal range monitor. The purge air is then exhausted back to the containment ventilation system exhaust duct.

4

~~See~~ figure 9.4-11 for the system arrangement drawing.

The accident range monitor flow path is provided with a particulate-iodine sample filter assembly. The sample filter, constructed of silver zeolite, has a collection efficiency greater than 90 percent for 0.3 micron diameter particles and for iodines. A bypass line is provided around the sample filter to permit filtered flow to continue to the high range monitor through the bulk filter. Sample filter removal is provided by means of quick disconnects. The sample filter is housed in a lead shield, mounted for ease of removal and replacement of filter media and capable of being transported to the onsite analysis facility during normal and accident conditions without the operator receiving doses in excess of those specified in

The ~~radiation~~ radioactivity detection assembly for the accident range monitor consists of a shielded chamber, a GM tube to monitor gross radioactivity (i.e. noble gases) with an accuracy of approximately 15% of logarithmic scale down to 40 Kev, and a check source mechanism.

The accident range monitor, particulate-iodine sample filter, and bulk filter are also provided with a purge assembly which can be manually initiated from the data aquisition module or either of the central control terminals after the accident range monitor, particulate-iodine sample filter, and bulk filter are isolated from the sample to permit purge air to flush the above equipment. The purge air is then exhausted back to the containment ventilation system exhaust duct.

Area monitors are provided for the normal and accident range monitors and for the sample filter assembly to compensate for fixed and variable background radiation [illegible].

6

The Data Aquisition Module (DAM) contains a microcomputer which performs background subtraction, applies conversion factors, and retains the data from each detector channel in history files consisting of the last 4 hours of ten-minute averages, the last 24 hours of one-hour averages and the last 24 days of one-day averages. Each DAM is AC operated with 8 hours of battery backup. Bi-directional communication is provided between the DAM and two central control terminals. Provisions exist to access each local DAM with a portable control terminal to conduct calibration and service functions at the DAM location. Each DAM with its detectors is optically isolated from the rest of the system. Failure of a DAM or its detector(s) will have no effect on any other portion of the system. Each DAM communicates with two (redundant) control terminals via two (redundant) communication interfaces. Because the local DAM is completely self supporting for the performance of ~~their~~ the tasks, even simultaneous power failures at both control terminals do not result in any loss of data accumulation or storage in the DAM.

The central control terminals are the operator's interface with the rest of the system. It is controlled with two keys, one for routine operation and the other (an "edit" key) for changing calibration parameters, alarm set points, etc. This provides two levels of system security. Each terminal has its own keyboard, printer and system status annuciator. Each control terminal includes a microcomputer which performs the functions of polling each local processor for operational status and data, logging any changes in status and associated data, logging history files automatically or upon keyboard request, performing calculations on data in the history files, annuciating status conditions, and communication error messages.

Changes in operating conditions are printed within seconds of the occurence. Data are presented only if the data are significant. History is printed in an interpretable, orderly manner, ensuring ease of operation. With a few entries on the keyboard, any data, status or parameters are presented.

An interface is provided to connect the radioactivity monitoring system to a ~~control~~ separate computer capable of determining off-site releases both during accident and recovery conditions.

4 The ~~radiation~~ radioactivity monitors are provided with check source mechanisms. Radionuclides for each monitor are chosen which best represent the radioactive ~~specie~~ isotope of interest. The check source mechanism can be either actuated at the DAM, ~~or~~ at the central control terminal in the control room, or at the central control terminal in the Technical Support Center.

The effluent radiation monitoring system ~~will be~~ is powered from ~~the~~ a non-interruptable power ~~supply~~ source.

The microprocessor based ~~radiation~~ radioactivity monitor ~~will~~ is also ~~be~~ capable of alarming the annunciators identified in the G.E. radiation monitoring ~~description~~ equipment, See subsection 11.5.2.2.4, in addition to the alarms indicated on the DAM and the central control terminals.

There are no seismic requirements for the containment ventilation ~~radiation~~ radioactivity monitoring system discussed herein. However, the system is designed to withstand local environmental conditions during and after an accident to assure system operability.

11.5.2.2.4.2 ~~Cmt Vent~~ Containment Ventilation G.E. Radiation Monitoring System

The G.E. sample panel is provided with a pair of sample filters (one for particulate collection and one for halogen collection) in parallel (with respect to flow) with a continuous gross radiation detection assembly. The gross radiation detection assembly consists of a shielded chamber, a beta-sensitive GM tube, and a check source. A radiation monitor in the control room analyzes and visually displays the measured gross radiation level. See figure 7.6-1b for the system arrangement drawing.

The sample panel shielded chambers can be purged with room air to check detector response to background radiation by using a three-way solenoid valve operated from the control room. The sample panel measures and indicates sample line flow. A solenoid operated check source operated from the control room can be used to check operability of the gross radiation channel.

Power is supplied from 125 V dc bus D for the radiation monitor and recorder, and from a 120 V ac local bus for the sample panel. The recorder has two pens, one used by this system and the other used by the offgas and radwaste building ventilation radiation monitoring system.

The radiation monitor has three trip circuits: two upscale (high-high and high) and one downscale (low). Each trip is visually displayed on the radiation monitor. These three trips actuate corresponding control room annunciators: containment ventilation high-high radiation, containment ventilation high radiation, and containment ventilation downscale. High or low sample flow measured at the sample panel actuates a control room containment ventilation sample high-low flow annunciator.

As discussed in subsection 11.5.2.2.4.1 the G.E. radiation monitoring system receives its sample from the microprocessor's flow monitoring and isokinetic sampler unit through the isokinetic sample panel.

11.5.2.2.5.3 Component Cooling Water Radiation Monitoring System

This system has a single channel for monitoring downstream of equipment in the component cooling water system.

11.5.2.2.6 Offgas and Radwaste Building Ventilation Radiation Monitoring System

This system monitors the offgas and radwaste building ventilation discharge, including radwaste storage tank vents, for gross radiation level and collects halogen and particulate samples. The system is identical to the containment ventilation radiation monitoring system with corresponding annunciators. In addition, a high-high radiation trip shuts down the radwaste building exhaust fans.

See new writeup

11.5.2.2.7 Fuel Handling Area Ventilation Radiation Monitoring System

This system monitors the fuel handling area ventilation radiation monitoring system discharge, including auxiliary building and fuel pool sweep vents, for gross radiation level and collects halogen and particulate samples. The system is identical to the containment ventilation radiation monitoring system with corresponding annunciators, except that power is from 125 V dc bus R. S

11.5.2.2.8 Turbine Building Ventilation Radiation Monitoring System

This system monitors the turbine building ventilation discharge for gross radiation level and collects halogen and particulate samples. The system is identical to the containment ventilation radiation monitoring system with corresponding annunciators except that power is from 125 V dc bus E. A two-pen recorder is shared between this system and the fuel handling area ventilation radiation monitoring system.

11.5.2.3 Inspection, Calibration and Maintenance

11.5.2.3.1 Inspection and Tests

During reactor operation, daily checks of system operability are made by observing channel behavior. At periodic intervals during reactor operation, the detector response (of each monitor provided with a remotely positioned check source) will be recorded together with the instrument background count rate to ensure proper functioning of the monitors. Any detector whose response cannot be verified by observation during normal operation or by using the remotely positioned check source will have its response checked with a portable check source. A record will be maintained showing the background radiation level and the detector response.

~~Radiation~~ Radioactivity Monitoring System

Microprocessor Based

11.5.2.2.6.1 ~~10~~ Offgas and Radwaste Building Ventilation ~~Radiation~~ Radioactivity Monitoring System

This system monitors the offgas and radwaste ~~bldg.~~ building ventilator discharge, including the radwaste storage tank vents for ~~gross radiation level~~ noble gasses, iodines, and particulates and collects halogen and particulate samples. This system is identical to the microprocessor based containment ventilation ~~radiation~~ radioactivity monitoring system discussed in Subsection 11.5.2.2 with corresponding annunciators. In addition a high-high radiation trip shuts down the radwaste bldg. exhaust fans. For the system arrangement see figures 9.4-4 and 5.

11.5.2.2.6.2 G.E. Offgas and Radwaste ~~Bldg.~~ Building Ventilation Radiation Monitoring System

This system monitors the offgas and radwaste building ventilation discharge, including radwaste storage tank vents, for gross radiation level and collects halogen and particulate samples. The system is identical to the GE containment ventilation radiation monitoring system with corresponding annunciators. In addition, a high-high radiation trip shuts down the radwaste building exhaust fans.

For a description of the G.E. system see subsection 11.5.2.2.4.2. For the system arrangement see figure 7.6-1B.

11.5.2.2.7 Fuel Handling Area Ventilation ~~Radiation~~ Radioactivity Monitoring System

11.5.2.2.7.1 Microprocessor Based Fuel Handling Area Ventilation ~~Radiation~~ Radioactivity Monitoring System.

This system monitors the fuel handling area ventilation discharge, including auxiliary ~~bldg.~~ building and fuel pool sweep vents for noble gasses, iodines, and particulates and collects halogen and particulate samples. This system is identical to the microprocessor based containment ~~radiation~~ radioactivity monitoring system discussed in subsection 11.5.2.2.4.1 with corresponding annunciators. For the system arrangement see figure 9.4-2.

11.5.2.2.7.2 G.E. Fuel Handling Area Ventilation Radiation Monitoring System

This system monitors the fuel handling area ventilation radiation monitoring system discharge, including auxiliary building and fuel pool sweep vents, for gross radiation level and collects halogen and particulate samples. The system is identical to the GE containment ventilation radiation monitoring system with corresponding annunciators.

The Fuel Handling Area Ventilation System ~~will~~ is powered from a ~~115~~ 120 V ac local BUS for the sample panel and 125V dc BUS B for the G.E. radiation monitor and recorder. See sub section 11.5.2.2.4.2 for a description of the [redacted] system. See Figure 7.6-1C for the system arrangement.

11.5.2.2.8 Turbine Building Ventilation ~~Radiation~~ Radioactivity Monitoring System

11.5.2.2.8.1 Microprocessor Based Turbine ~~Bldg~~ Building Ventilation ~~Radiation~~ Radioactivity Monitoring System

This system monitors the turbine bldg. ventilation discharge for ~~gross radiation level,~~ noble gasses iodines, particulates and collects halogen and particulate samples. This system is identical to the microprocessor based containment ~~radiation~~ radioactivity monitoring system discussed in Subsection 11.5.2.2 with corresponding annunciators. For the system arrangement see figures 9.4-6 and 7.

11.5.2.2.8.2 G.E. Turbine Building Ventilation Radiation Monitoring System

This system monitors the turbine building ventilation discharge for gross radiation level and collects halogen and particulate samples. The system is identical to the GE containment ventilation radiation monitoring system with corresponding annunciators.

The Turbine ~~Bldg~~ Building Ventilation system ~~will~~ is ~~the~~ powered from a 120 V ac local BUS for the sample panel and 125 V dc BUS B for the G.E. radiation monitor. A two pen recorder is shared between this system and the Fuel Handling Area Ventilation G.E. radiation monitoring system. See subsection 11.5.2.2.4.3 for a description of the system. See figure 7.6-15 for the system arrangement.

New section - Insert after ~~[illegible]~~ 11.5.2.2.8

11.5.2.2.9 Standby Gas Treatment A & B Exhaust Ventilation ~~Radiation~~ Radioactivity Monitoring Systems

These systems monitor the Standby Gas Treatment System (SGTS) A & B discharges for noble gas, iodine, ~~and particulates~~, and collect halogen and particulate samples. These systems are identical to the containment ventilation microprocessor based ~~radiation~~ radioactivity monitoring system discussed in subsection 11.5.2.2.4.1 with the following exceptions: (For Sys ~~Arrange~~ See Figure 6.5-2

1) The SGTS normally operates during accident conditions; therefore, the SGTS ~~radiation~~ Radioactivity monitoring system will operate during accident and recovery conditions.
2) Each of the SGTS ~~effluent~~ ~~radiation~~ radioactivity monitoring systems will be manually initiated by the operator. ~~Automatic~~ Initiation of the ~~radiation~~ radioactivity monitoring system will automatically start the isokinetic sampling portion of the system.
3) The SGTS ~~radiation~~ radioactivity monitoring systems ~~will~~ does not have an associated G.E. system*.

4. Any portion of the SGTS effluent ~~radiation~~ radioactivity monitoring systems which penetrates the boundry of the SGTS ~~will be~~ is designed to the seismic ~~and~~ criteria of the exhaust duct or ~~vent it is attached to~~.

5. The SGTS ~~radiation~~ radioactivity monitoring system ~~will~~ annunciate ~~only~~ at the Data Aquisition Module, and at both ~~the~~ central control terminals.

The system has electronic testing and calibrating equipment which permits channel testing without relocating or dismounting channel components. An internal trip test circuit, adjustable over the full range of the readout meter, is used for testing. Each channel is tested at least semiannually prior to performing a calibration check. Verification of valve operation, ventilation diversion, or other trip function will be done at this time if it can be done without jeopardizing the plant safety. The tests will documented.

11.5.2.3.1.1 Detailed Inspection and Tests

a. The following monitors have alarm trip circuits which can be tested by using test signals or portable gamma sources:

   1. Main steam line
   2. Containment and drywell ventilation exhaust
   3. Auxiliary building fuel handling area
   4. Auxiliary building fuel handling area pool sweep
   5. Offgas pretreatment
   6. Carbon bed vault

b. The following monitors include built-in check sources and purge systems which can be operated from the control room:

   1. Offgas post-treatment
   2. Containment ventilation
   3. Offgas and radwaste building
   4. Fuel handling area ventilation
   5. Turbine building ventilation
   6. Standby Gas Treatmen System A & B

c. The following monitors include built-in check sources which can be operated from the control room:

   1. Radwaste effluent
   2. Standby service water
   3. Component cooling water

11.5.2.3.2 Calibration

The continuous radiation monitor's calibration is traceable to certified National Bureau of Standards or commercial radionuclide standards, and is accurate to at least ± 15 percent. The source-detector geometry during primary calibration is identical to the sample-detector geometry in actual use. Secondary standards which were counted in reproducible geometry during the primary calibration are supplied with each continuous monitor for calibration after installation. Each continuous monitor is calibrated annually during plant operation or during the refueling outage if the detector is not readily accessible. A calibration can also be performed by using liquid or gaseous radionuclide standards or by analyzing particulate, Iodine or gaseous grab samples with laboratory instruments.

11.5.2.3.2.1 Specific calibration criteria are as follows:

a. The following monitor shall have as a criterion for calibration response to a gross gamma signal with the calibration factor in mr/hr per μCi/sec being derived from periodic analyses of grab samples:

   1. Offgas pretreatment

b. The following monitors shall have as a criterion for calibration response to a gross gamma signal with the calibration factor in counts/min per μCi/sec being derived from periodic analyses of grab or filter samples:

   1. Offgas post-treatment

   2. Containment ventilation

   3. Offgas and radwaste building vent

   4. Fuel handling area vent

   5. Turbine building vent

   6. Standby Gas Treatment System A & B

   7 ~~6~~. Radwaste effluent

   8 ~~7~~. Standby service water

   9 ~~8~~. Component cooling water

c. The following monitors shall be calibrated to read the gross gamma dose rate in mr/hr:

   1. Main steam line

   2. Containment and drywell vent

3. Auxiliary building fuel handling area

4. Auxiliary building fuel handling area pool sweep

5. Carbon bed vault

11.5.2.3.3 Maintenance

The channel detector, electronics and recorder are serviced and maintained on an annual basis or in accordance with manufacturers' recommendations to ensure reliable operations. Such maintenance includes cleaning, lubrication, and assurance of free movement of the recorder in addition to the replacement or adjustment of any components required after performing a test or calibration check. If any work is performed which would affect the calibration, a recalibration is performed at the completion of the work.

11.5.2.3.4 Audits and Verifications

Independent audits and verifications of test, calibration and maintenance records and procedures are conducted as described in Section 17.2.

11.5.3 Effluent Monitoring and Sampling

11.5.3.1 Implementation of General Design Criterion 64

All potentially radioactive effluent discharge paths are continuously monitored for gross radiation level. Liquid releases are monitored for gross gamma. Solid waste shipping containers are monitored with gamma sensitive portable survey instruments. Gaseous releases are monitored for gross gamma. The following gaseous effluent paths are sampled and monitored:

a. Containment Ventilation System

b. Offgas and Radwaste Building Ventilation System which includes the offgas system and the storage tank vents.

c. Fuel Handling Area Ventilation System which includes the auxiliary building, fuel handling area, and fuel pool sweep ventilation systems

d. Turbine Building Ventilation System

e. Standby Gas Treatment System

The following liquid effluent path is sampled and monitored:

Liquid Radwaste System

All monitors have wide ranges and are listed in Table 11.5-1.

TABLE 11.5-1 (Cont.)

PROCESS AND EFFLUENT RADIOACTIVITY MONITORING SYSTEMS

MONITORED PROCESS	NO. OF CHANNELS	DETECTOR TYPE	SAMPLE LINE or DETECTOR LOCATION	CHANNEL RANGE	UPSCALE SET POINT WARNING ALARM	TRIP	SCALE	PURPOSE OF MEASUREMENT	PRINCIPAL RADIONUCLIDES MEASURED
Containment Ventilation GE SYSTEM	1	Geiger-Muller Tubes	Sample line	10 to 10^6 counts/min	1×10^3	not applicable	5 dec. log	Audit discharge to environs	Xe-133, Kr-85
Offgas and Radwaste Building Vent GE SYSTEM	1	Geiger-Muller Tube	Sample line	10 to 10^6 counts/min	5×10^5	1×10^6	5 dec. log	Audit discharge to environs	Xe-133, Kr-85
Fuel Handling Area Vent GE SYSTEM	1	Geiger-Muller Tube	Sample line	10 to 10^6 counts/min	1×10^3	not applicable	5 dec. log	Audit discharge to environs	Xe-133, Kr-85
Turbine Building Vent GE SYSTEM	1	Geiger-Muller Tube	Sample line	10 to 10^6 counts/min	1×10^3	not applicable	5 dec. log	Audit discharge to environs	Xe-133, Kr-85
Cont Vent	1	Scintillation Detector	"	$10-10^6$ counts/min	1×10^3 cpm	NOT APPLICABLE	Later	"	I-131
Microprocessor SYS. (SEE NOTE 1)	1	Scintillation Detector	"	$10-10^6$ counts/min	1×10^3 cpm	not APPLICABLE		"	CS-137
	1	Scintillation Detector	"	$10^{-7} - 6 \times 10^{-2}$ µCi/cc	later	NOT APPLICABLE		"	Xe-133, Kr-85
	1	GM TUBE	"	$2 \times 10^{-2} - 4 \times 10^{2}$ µCi/cc	"	"		"	" "
	1	GM TUBE	"	$10^{-4} - 10^{1}$ µCi/cc	"	"		"	" "
	1	GM TUBE	"	$10^{1} - 10^{5}$ µCi/cc	"	"		"	" "
Offgas & Radwaste Bldg Vent.	1	Scintillation Detector	"	$10-10^6$ cpm	1×10^3 cpm	"		"	I-131
Microprocessor System	1	Scintillation Detector	"	$10-10^6$ cpm	1×10^3 cpm	"		"	CS-137
	1	Scintillation Detector	"	$10^{-7} - 6 \times 10^{-2}$ µCi/cc	later	"		"	Xe-133, Kr-85
	1	GM TUBE	"	$2 \times 10^{-2} - 4 \times 10^{2}$ µCi/cc	"	"		"	" "
	1	GM TUBE	"	$10^{-4} - 10^{1}$ µCi/cc	"	later		"	" "
	1	GM TUBE	"	$10^{1} - 10^{5}$ µCi/cc	"	later		"	" "

NOTES:

1. Typical for FHA Ventilation, Turbine Bldg. Ventilation, and Standby Gas Treatment System A & B.

Table already submitted to licensing with response to Q 330.23 (CN #1168)

TABLE 11.5-3

PROVISIONS FOR MONITORING AND SAMPLING
GASEOUS AND LIQUID STREAMS

Process System	Monitor Provisions				Sample Provisions		
	In Process		In Effluent		In Process	In Effluent	
	Cont.[1]	ACF[2]	ACF[2]	Cont.[1]	Grab[3]	Grab[3]	Cont.[1]
A. Gaseous Streams							
Offgas posttreatment	NG	NG			NIGR		
Offgas pretreatment (Condenser Air Removal)	NG				NIG		
Containment ventilation system[4]	NG	NG		NG I		NIGRT	NGI
Offgas & RW bldg. vent. system[5]			NG	NG I		NIGRT	NGI
Fuel-handling area vent. system[6]	NG	NG		NG I		NIGRT	NGI
Turbine bldg. vent. system[7]				NG I		NIGRT	NGI
STANDBY GAS TREATMENT SYSTEM A & B				NG I		NIGRT	NGI
Carbon bed vault	NG				IG		
B. Liquid Streams							
Floor drain sample tanks[8]						GR	
Equip. drain sample tanks[8]						GR	
Chemical waste distillate sample tanks[8]					G	GR	
Condensate storage tank					GR		
Laundry waste monitoring tank[9]					GR		
Refueling water storage tank					GR		
Condensate storage tank dike sump[10]					G		
Liquid radwaste effluent			G	G		GRT	
Component cooling water system	G				G		
Standby service water system	G				G		

320.20

7.1.2.1.40 In-Containment Area Radiation Monitoring System - Instrumentation and Controls

As a result of TMI Lessons Learned, the applicant has installed a high range in-containment radiation monitoring system which will be operable before January 1, 1982.

7.1.2.1.40.1 Safety Design Basis

The monitoring of high range gamma radiation levels in the containment and the drywell after a design basis loss-of-coolant accident is accomplished by the area radiation monitoring system.

The design basis for this system is discussed in subsection 12.3.4.3.

7.1.2.1.40.2 Specific Regulatory Requirements

Specific regulatory requirements met by the area radiation monitoring system are listed under Safety Related Display Instrumentation in Table 7.1-3.

7.5.1.2.3.6 In Containment Area Radiation

Two high range containment radiation and two high range drywell radiation signals are transmitted from four separate radiation detectors to four control room ~~indicators~~ monitors and two ~~two~~ 2-pen recorders. One pen records ~~the~~ containment radiation and one pen records drywell radiation on each of the two independent recorders. Two radiation detectors are located at opposite sides of the containment and two detectors are located at opposite sides of the drywell. All detectors are accessible for maintenance, removal for calibration, or replacement when entry to the [illegible]. Monitors are qualified to function under loss-of-coolant accident condition.

over a range of 1 to 10^7 rads/hr

The energy dependence of the system is 60 KeV to 3 MeV photons with ±8 percent accuracy for photons of 0.1 to 3.0 MeV.

The power sources for the four channels are fed from two seperate ESF ac buses.

TABLE 7.5-1

SAFETY-RELATED DISPLAY INSTRUMENTATION

System	Parameter	Type of Readout	Number of Channels	Range	Accuracy	Readout Location	Post-Accident Tracking	Class 1E Power
Rod control and information system	Control rod position	Lights	2/Rod	NA	NA	CR	No	No
	Control rod scram valves	Lights	1/Valve	NA	NA	CR	No	No
	Bypass or inoperable status	Lights	(1)	NA		CR	No	(1)
Neutron Monitoring	Power range neutron flux	Recorder	4	0-125%	±5% FS	CR	No	No
	Bypass or inoperable	Lights	(1)	NA	NA	CR	No	(1)
Nuclear boiler	Reactor vessel pressure	Recorder	2	0-1500 psig	±2% FS	CR	Yes	Yes
	Reactor vessel water level	Recorder	2	-150"/0/+60"	±2% FS	CR	Yes	Yes
Containment and drywell	Drywell pressure	Recorder	2	0-30 psig	±2% FS	CR	Yes	Yes
	Containment pressure	Recorder	2	0-15 psig	±2% FS	CR	Yes	Yes
	Drywell & CRD cavity temperature	Recorder	4	0-400 F	±2% FS	CR	Yes	Yes
	Containment & drywell temperature	Multipoint Recorder	2 (8 ctmt. & 4 drwl. locations each)	0-400 F	±2% FS	CR	No	No
	Suppression pool level	Recorder	2	14-26 ft	±2% FS	CR	Yes	Yes
	Suppression pool level	Recorder	2	18-19 ft	±2% FS	CR	No	Yes
	Suppression pool temperature	Recorder	4 (6 locations each)	0-200 F	±3% FS	CR	Yes	Yes
	Isolation valve position	Lights	1 per valve	NA	NA	CR	No	[illegible]
	Bypass or inoperable status: includes: CRVICS; Aux. bldg. isolation; FWLCS; Suppression pool temperature monitoring		2	NA	NA	CR	No	Yes
	In-Containment Area Radiation	meter and Recorder	4 (2 ctmt & 2 drywell)	1-10^7 R/hr	±8%	CR	Yes	Yes

Sheet 1 of 5

12.3.4.3 In-Containment Area Radiation Monitoring

The ICARM system is provided to detect, indicate, and reco gamma radiation levels in the containment and the drywell over a range of 1 to 10^7 R/hr ~~[illegible]~~ during and following a loss-of-coolant accident. The system meets the design and qualification criteria as outlined in IEEE 323-1974, and IEEE 344-1975.

The ICARM system has no function related to the safe shutdown of the plant, or to the quantitative monitoring of releases of radioactive material to the environment. The system's only purpose is to monitor high range radiation levels in the containment and drywell after an accident.

12.3.4.3.1 Criteria for Monitor Selection

The following design criteria are applicable to the ICARM system:

Rangeability - Seven decades of range are available on each monitor.

Sensitivity - The ICARM is gamma sensitive to photon energies of 60 KeV to 3 MeV.

Response - System response time is less than 1 second.

Energy Dependence - The dose rate (mR/hr) reading is within 8 percent of the actual dose rate in each detected area from photon energies between 0.1 MeV and 3.0 MeV

Environmental Dependence The system ~~meets the requirements for all~~ variations of temperature, pressure and relative humidity within each area monitored. Qualification includes 100 percent relative humidity and temperatures between 32F and 357F.

Exposure Life Each detector maintains its characteristics up to an integrated gamma dose of 2 x 10 rads

12.3.4.3.2 Criteria for Location of Monitors

Two radiation detectors are mounted at north and south locations inside the containment wall and two are mounted ~~detectors~~ at east and west locations inside the [illegible] wall.

The detector locations are given on the radiation zoning and access control drawings, Figures 12.3-15 and 12.3-17.

12.3.4.3.3 System Description (ICARM)

The ICARM system continuously detects, ~~&~~ indicates and records high-range gamma radiation levels in the containment and the drywell during and following an accident.

Each radiation monitoring channel consists of a local ion chamber detector, a control room indicator trip unit, and a control room recorder. Each monitor is provided with radiation and circuit failure alarm indicating lights.

In addition, each monitor is provided with a seven decade logarithmic meter which reads in R/hr.

Each monitor is provided with an electronic check source test which is automatically initiated every 17 minutes to assure the integrity of the electrical configuration and electrical operation of the detector/cable/readout system. Only a successful check turns on a front panel indicating light until the next check to provide ~~[illegible]~~ indication.

proper operation of the system. An unsuccessful check initiates a control room failure ~~alarm~~ light on the front of the trip unit.

All electronics are solid-state.

All monitors are independent and failure of one monitor has no effect on any other.

The radiation monitors are powered from a class 1E power supply. Emergency power to this bus is provided by the station battery through an inverter.

System characteristics are given in Tables 12.3-6 and 12.3-7

12.3.4.3.4 Safety Evaluation

The ICARM system is qualified to function ~~in~~ a loss-of-coolant accident environment. The system serves to monitor high range radiation levels in containment and drywell during accident conditions. The system is not essential for safe shutdown of the plant.

The system is designed to operate unattended while detecting, indicating and recording high range gamma radiation, primarily post accident. Radiation dose rate from ~~at~~ each detector is indicated and recorded remotely in the control room. The trip unit circuitry ~~monitors~~ ~~cause~~ is such that an alarm light will be ~~energized~~ illuminated on ~~the monitor~~ if radiation levels exceed preset limits or if the system malfunctions or loses its power source.

* The manufacturer has stated that the equipment supplied does not utilize a radioactive check source for in-situ calibration. Instead,

12.3.4.3.5 Calibration and Testing

Each of the monitors is calibrated at 12, 35, and 300 R/hr by the instrument manufacturer prior to shipment using standards certified by the National Bureau of Standards (NBS) or traceable to NBS.* during each refueling outage the instruments will be returned to the manufacturer for recalibration at 12, 35 and 300R/hr, to an accuracy defined by SAMA Standard PMC 20. With each calibration, the manufacturer shall provide written certification that each area radiation monitoring system has been calibrated within the specified accuracy required. Upon installation, the instruments shall be tested for calibrated accuracy by electronic signal substitution.

The system includes an internal test trip circuit. The test signal is electrically fed into the unit so that a meter reading is provided in addition to a real trip. The test control is located on the front of the indicator trip unit. In addition, an automatic electronic "check source" test provides visual control room indication on system malfunction.

TABLE 12.3-6

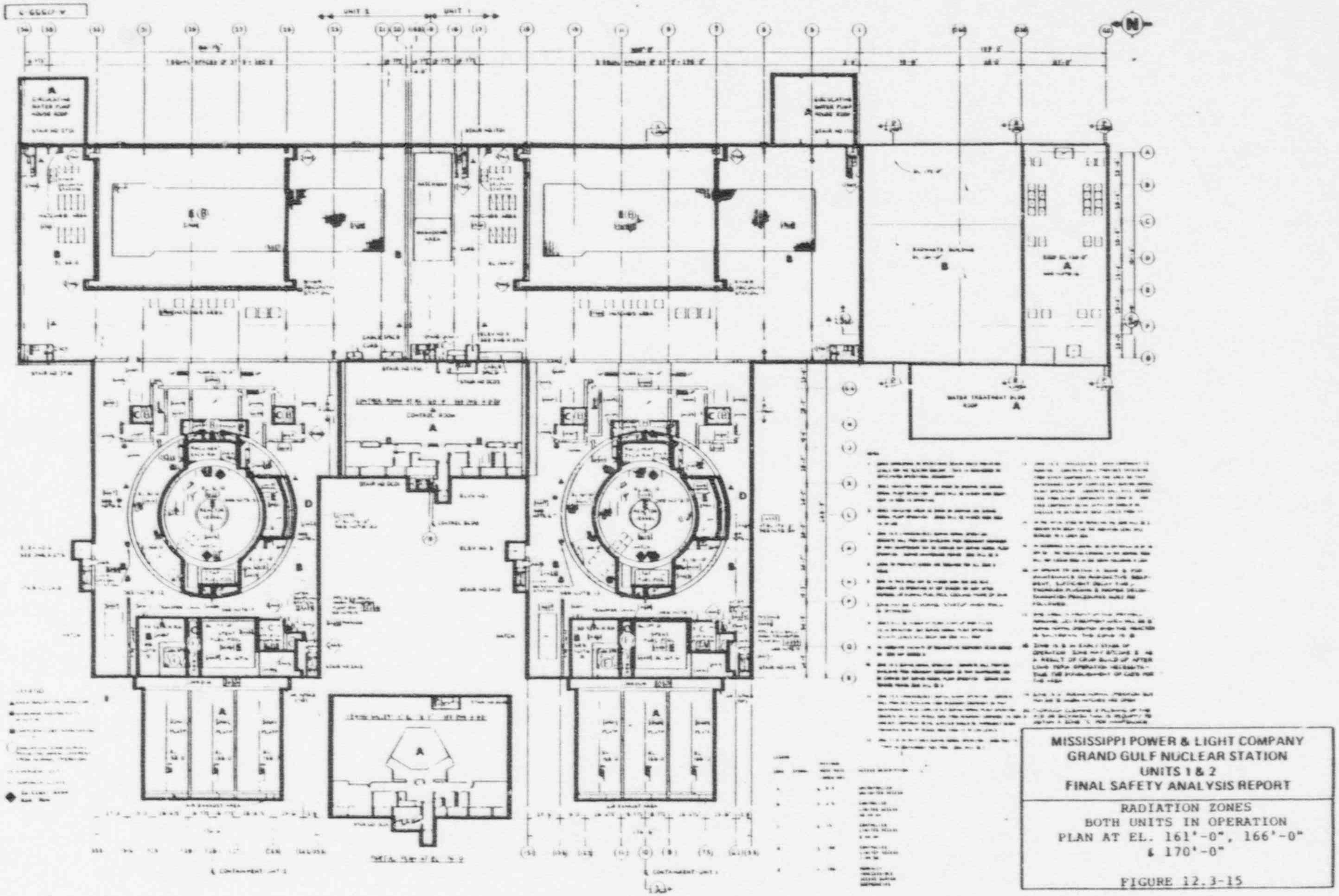
In Containment Area Radiation Monitors

Location	Elevation (ft.)	Range (R/hr)
CTMT -North	208	1-10^7
CTMT South	208	1-10^7
DRYWELL East	166	1-10^7
DRYWELL West	166	1-10^7

TABLE 12.3-~~2~~ 7

IN-CONTAINMENT AREA RADIATION MONITORING EQUIPMENT CHARACTERISTICS

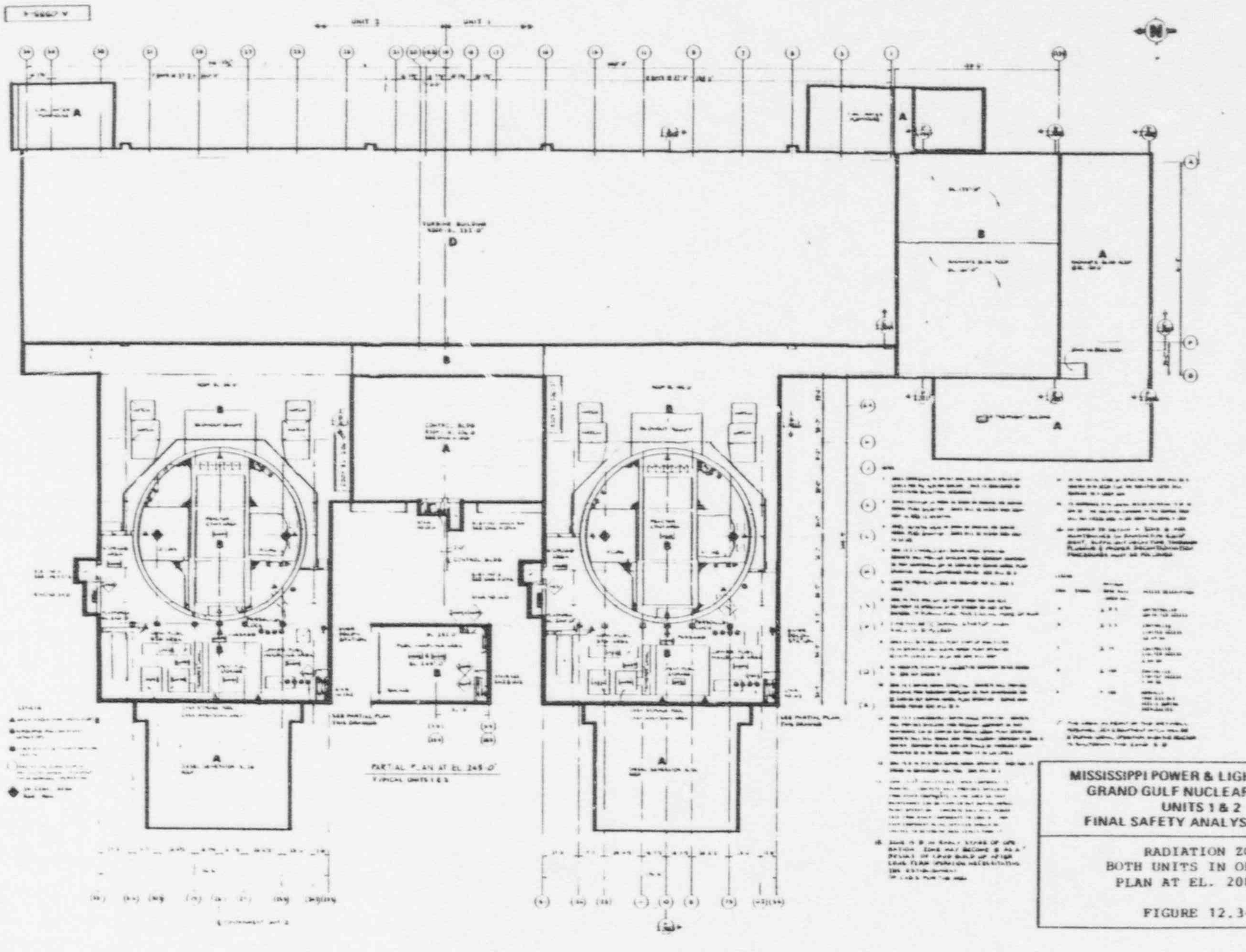
Parameter	Characteristics
Detector assembly	
Type	Gamma (Nitrogen-filled Ion Chamber)
Quantity	2 Containment 2 Drywell
Mounting	Wall
Range	1.0 to 10^7 R/hr
Energy dependence	±8% from 0.1 MeV to 3.0 MeV
~~Local~~ Control Room Alarm Unit	
~~Alarm~~ HIGH RADIATION	Alarm indicating light on monitor
~~Mounting~~ FAILURE	Alarm indicating light on monitor
Indicator Trip Unit	
Range	1.0 to 10^7 R/hr
Indicator	4-in scale panel meter
Indicator accuracy	±5% around the midpoint of each decade
Overall Channel Accuracy	±8%



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RADIATION ZONES
BOTH UNITS IN OPERATION
PLAN AT EL. 161'-0", 166'-0" & 170'-0"

FIGURE 12.3-15



MISSISSIPPI POWER & LIGHT COMPANY
GRAND GULF NUCLEAR STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RADIATION ZONES
BOTH UNITS IN OPERATION
PLAN AT EL. 208'-10"

FIGURE 12.3-17

031.74

7.5.1.2.2.2 Reactor Pressure

Two reactor pressure signals are transmitted from two independent pressure transmitters and are recorded on two, two-pen recorders which are operable before and after a Safe Shutdown Earthquake (SSE). One pen records pressure and the other pen records the wide range level. The range of recorded pressure is from 0 to 1500 psig. Power sources are as stated in subsection 7.5.1.2.2.1.

7.5.1.2.2.3 Reactor Isolation

The reactor operator may verify reactor isolation by observing one or more of the following indications:

a. Isolation valve position lamps indicating valve closure. See Figure 7.5-2. The power source is the same as for the associated valve motor-operator.

b. Main steam line flow indication downscale. See Figure 7.5-1. The power source is instrument ac from one of the standby ac buses.

031.74

7.5.1.2.3 Containment and Drywell

INSERT 1

7.5.1.2.3.1 Containment and Drywell Pressure

~~Two wide-range containment pressure and two wide-range drywell pressure signals are transmitted from four separate pressure transmitters and are recorded on two two-pen recorders in the control room. One pen records the containment pressure, and the other pen records the drywell pressure on each of the two independent recorders.~~ The power sources for the two channels each of containment and drywell pressure are from the two ESF dc buses through two independent inverters. The containment and drywell pressure monitoring instrumentation is shown in Figure 7.5-5. 031.74

INSERT 1

7.5.1.2.3.1 Containment and Drywell Pressure

Two wide-range containment pressure, two narrow-range containment pressure and two wide-range drywell pressure signals are transmitted from six separate pressure transmitters and are continuously recorded and displayed on two three-pen recorders in the control room. One pen records the wide-range containment pressure, one pen records the narrow-range containment pressure, and one pen records the wide-range drywell pressure on each of the two independent recorders. The containment and drywell pressure indication is exclusively for operator information and does not perform a control function. This pressure indication meets the design and qualification criteria for accident monitoring instrumentation as outlined in Regulatory Guide 1.97. The power sources . . .

INSERT 1A page 2

The total response time for these pressure monitoring systems is in the order of 1-2 seconds, which is adequate to detect and record any significant pressure impulses. The accuracy of the record as shown in Table 7.5-1, is sufficient to provide the operator with an adequate indication of drywell and containment pressure.

insert to pag. 1

Insert A

All six pens record simultaneously, thus providing a constant indication of the full range of pressure monitoring.

TABLE 7.5-1

SAFETY-RELATED DISPLAY INSTRUMENTATION

System	Parameter	Type of Readout	Number of Channels	Range	Accuracy	Readout Location	Post-Accident Tracking	Class 1E Power
Rod control and information system	Control rod position	Lights	2/Rod	NA	NA	CR	No	No
	Control rod scram valves	Lights	1/Valve	NA	NA	CR	No	No
	Bypass or inoperable status	Lights	(1)	NA		CR	No	(1)
Neutron Monitoring	Power range neutron flux	Recorder	4	0-125%	±5% FS	CR	No	No
	Bypass or inoperable	Lights	(1)	NA	NA	CR	No	(1)
Nuclear boiler	Reactor vessel pressure	Recorder	2	0-1500 psig	±2% FS	CR	Yes	Yes
	Reactor vessel water level	Recorder	2	-150"/0/+60"	±2% FS	CR	Yes	Yes
Containment and drywell	Drywell pressure	Recorder	2	-10/0/+40 psig	± 1.3% FS	CR	Yes	Yes
	Containment pressure	Recorder	4	5/0/+5 psig	± 1.3% FS	CR	Yes	Yes
	Drywell & CRD cavity temperature	Recorder	4	0-50 psig 0-600 F	± 1.3% FS	CR	Yes	Yes
	Containment Temperature	Recorder	4	0-400 F	± 1.3% FS	CR	Yes	Yes
	Containment & drywell temperature	Multi-point Recorder	2 (6 cmt. & 4 drwl. locations each)	0-400 F	± 0.6% FS	CR	No	No
	Suppression pool level	Recorder	2	12'4"-24'4"	± 1.3% FS	CR	Yes	Yes
	Suppression pool level	Recorder	2	18-19 ft	± 1.3% FS	CR	No	Yes
	Suppression pool temperature	Recorder	4 (6 locations each)	30-230 F	± 1.3% FS	CR	Yes	Yes
	Isolation valve position	Lights	1 per valve	NA	NA	CR	No	No
	Bypass or inoperable status: includes: CRVICS Aux. bldg. isolation FWLCS Suppression pool temperature monitoring		2	NA	NA	CR	No	Yes

7.5.1.2.3.2 Containment and Drywell Temperature

Containment and drywell air temperatures are monitored at 32 locations in all quadrants and at various elevations as shown in Figure 7.5-5. These temperatures are recorded on two multipoint and four ~~two~~ three-pen recorders in the control room. The temperature instrumentation is divided into two separate and independent channels, each consisting of 16 temperature monitoring sensors, one multipoint recorder, and two ~~two~~ three-pen recorders. The power source for the two channels are fed from the two ESF ac buses. 031.74

7.5.1.2.3.3 Suppression Pool Level

Two wide-range and two narrow-range suppression pool level signals ~~are transmitted from four~~ Continuously separate differential pressure type ~~level transmitters and are~~ recorded on two two-pen recorders in the control room. One pen records the wide-range level, and the other pen records the narrow-range level on each of the two independent ~~recorders.~~ The power sources fro the two channels each
INSERT 1 of narrow- and wide-range suppression pool level are from the two ESF dc buses through two independent inverters. 031.74

7.5.1.2.3.4 Suppression Pool Temperature

Suppression pool temperature is measured by six sensor groups of four channels each. Two of the channels are recorded on four three-pen recorders in the control room, and are used for post-LOCA monitoring. The power sources for these two recorded channels are fed from the two ESF ac buses. The suppression pool temperature monitoring instrumentation is discussed in detail in subsection 7.6.1.11. 2

7.5.1.2.3.5 Containment and Drywell Isolation

The reactor operator may verify containment and drywell isolation by observing the following indication: 031.74

a. Isolation valve position lamps indicating valve closure. See Figures 7.5-2 and 7.5-3. The power sources are the same as for the associated valve motor-operators or solenoid pilot valves. 031.74

Amend. 41 7/80

INSERT 1 The wide-range water level indicators monitor the suppression pool level from the top of the upper most suppression pool vent to approximately 5'9" above the normal water level. Once the water level falls below the lower reference, the suppression pool has lost its function and the exact level becomes irrelevant. The system accuracy, as shown in Table 7.5-1 is sufficient to provide the operator with an adequate indication of containment water level. The wide-range indication channels meet the design and qualification criteria as outlined in Regulatory Guide 1.97. The narrow-range channels meet the requirements of Regulatory Guide 1.89. ~~The power sources from the..~~

7.5.1.2.8.2 Hydrogen Control System

Operation of the hydrogen control system recombiners may be verified by observing the following indications:

a. Hydrogen recombiner operating lights indicating that the recombiners are energized. See Figure 7.5-3. The power sources are the same as for the associated recombiners.

b. Two meters, one displaying recombiners temperature and the other displaying recombiner heater power input, for each recombiner are located in the control room. Each recombiner is equipped with three thermocouples, any one of which may be selected to be displayed on the associated meter. See Figure 7.5-3. The power sources are fed from the same ESF buses as the hydrogen recombiners.

Insert 1

7.5.1.2.8.3 Containment and Drywell Hydrogen Monitoring

Two containment hydrogen concentration and two drywell hydrogen concentration signals are transmitted from four separate hydrogen analyzers and are recorded on two two-pen recorders in the control room. One pen records the containment hydrogen concentration on each of the two independent recorders. Insert 2 Each hydrogen analyzer uses a sample drawing system which removes the sample from the containment or drywell, as appropriate, and employs a thermal conductivity analyzer. The sample drawing points are located at widely separated and appropriately placed points inside the containment and drywell. Each hydrogen analyzer can be calibrated by introducing zero and span gasses into the sample system to the analyzer. Calibration is initiated and adjustments are made, if necessary, at the hydrogen analyzer sample panels in the auxiliary building. See Figure 7.5-3. Inserts 3 and 4 The power sources are fed from the two ESF ac buses. The containment and drywell hydrogen analyzing instrumentation is shown in Figure 6.2-81. 031.74

7.5.1.2.9 Standby Gas Treatment System (SGTS)

Operation of the SGTS may be verified by observing the following indications:

a. Charcoal filter train fan and enclosure building recirculation fan operating lights indicating running equipment. See Figure 7.5-3. The power sources are from the same ESF dc buses which provide control power

7.5.1.2.8.3

INSERT 1 ... and one pen records the drywell hydrogen concentration...

INSERT 2 Measurement capability is provided over the range of 0 to 10 percent hydrogen concentration under both positive and negative ambient pressu

INSERT 3 The system accuracy as shown in Table 17.15-1 is sufficient to provide the operator with an adequate indication of hydrogen concentration. The hydrogen monitoring system meets the design and qualification criteria as outlined in Regulatory Guide 1.97.

Insert 4 (after Insert 3)

The hydrogen monitoring and the low pressure ~~core~~ coolant injection are simultaneously automatically activated upon a loss-of-coolant accident, and within 60 seconds, the hydrogen analyzers are sufficiently warmed to continuously analyze and record containment and drywell hydrogen concentration.

TABLE 7.5-1 (Cont.)

System	Parameter	Type of Readout	Number of Channels	Range	Accuracy	Readout Location	Post-Accident Tracking	Class 1E Power	
Standby service water (cont.)	SSW valve positions valve	Lights	1 per	NA	NA	CR	No	Yes	
	Bypass or inoperable status	Lights	1 per loop	NA	NA	CR	No	Yes	
MSIVLCS	Steam line pressure (low)	Meter	6	30" Hg vacuum 10 psig	±2% FS	CR	No	Yes	
	Steam line pressure (high)	Meter	6	0-100 psig	±2% FS	CR	No	Yes	
	Bypass or inoperable status	Lights	2	NA	NA	CR	No	Yes	
FWLC	FWLC valve positions	Lights	1 per valve	NA	NA	CR	No	Yes	
Combustible gas control system	Drywell/containment differential pressure	Recorder	2	-10/0/+20 psid	±1.3% FS	CR	Yes	Yes	
	Drywell hydrogen	Recorder	2	0-10%	±/3% FS	CR	Yes	Yes	
	Containment hydrogen	Recorder	2	0-10%	±/3% FS	CR	Yes	Yes	
	Hydrogen recombiner temperature	Meter	1 per recombiner	0-2000 F	±4% FS	CR	No	Yes	
	Hydrogen recombiner power	Meter	1 per recombiner	0-100 KW	±3% FS	CR	No	Yes	28
	Drywell purge compressor on/off	Lights	1 per compressor	NA	NA	CR	No	Yes	
	Drywell purge valve position	Lights	1 per valve	NA	NA	CR	No	Yes	
	Hydrogen recombiner on/off	Lights	1 per recombiner	NA	NA	CR	No	Yes	
	Bypass or inoperable status	Lights	2	NA	NA	CR	No	Yes	
Standby gas treatment	Enclosure bldg/outside atmosphere differential pressure	Recorder	4	-1/0/+1 inches water	±3% FS	CR	No	Yes	
	Charcoal filter train flow	Recorder	1 per train	0-5000 cfm	±2% FS	CR	No	Yes	
	Charcoal filter differential pressure	Recorder	1 per filter	0-10 inches water	±2% FS	CR	No	Yes	