October 18, 1988

Docket No. 50-416

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See next page

LICENSEE: System Energy Resources, Inc. (SERI)

FACILITY: Grand Gulf Nuclear Station, Unit 1 (GGNS-1)

SUBJECT: SUMMARY OF AUGUST 31, 1988 MEETING REGARDING LICENSING ACTIVITIES

A meeting was held with System Energy Resources, Inc., on August 31, 1988 in Rockville, Maryland. The purpose of the meeting was to discuss the status of licensing activities and the schedule for completing them. Enclosure 1 is a list of attendees at the meeting. Enclosure 2 is the meeting agenda prepared by the NRC staff. Enclosure 3 is the licensee's proposed markup of changes to Technical Specifications (TS) in Section 6.0. Enclosure 4 is a handout prepared by the licensee for use in describing the design and operation of the proposed new alternate decay heat removal system to be installed in the third refueling outage (RFO3) scheduled to begin March 1, 1989.

The staff provided the status of the licensee's submittals requesting TS changes. To achieve greater operational flexibility, the licensee has proposed to separate the diesel generator 24-hour surveillance test from the test of a simulated loss-of-coolant accident (LOCA) and simulated loss of offsite power (LOP). This is a generic TS change that is being reviewed by the Electrical Systems Branch. The present TS require running the LOCA and LOP test immediately after the 24-hour test when diesel generator thermal conditions have stabilized. A footnote to the TS allows a separate LOCA and LOP test if there is a failure of this test after a successful 24-hour test. The footnote requires a prior run of the diesel generator for one hour or until thermal conditions have stabilized. The proposed separate LOCA and LOP test would have the same prerequisite as the present footnote. The staff indicated that this is priority 4 review and the target date for issuance of the amendment is January 15, 1989.

The proposed change to the snubber sampling plans in the TS is a priority 4 review. The proposal is similar to those reviewed and accepted on other plants. The amendment is scheduled for issuance on January 30, 1989.

The proposal to lower the downtravel power cutoff setpoint for the fuel hoist is a priority 2 review and is scheduled for completion on December 15, 1988. The basis for the cutoff point is to allow the fuel grapple to travel low enough to engage fuel assembly and control blade guide handles and to prevent the grapple from traveling below the reactor vessel top grid where it may be damaged. The present TS uses fuel assembly handles as the reference for making the setpoint, but fuel assembly length increases with irradiation, and the licensee experienced difficulty during RFO2 engaging control blade guides which do not grow with irradiation. The new TS would use the reactor vessel top guide as a reference for making the setpoint.

DF01 1/1

The proposed changes to TS 3.0 and 4.0 (Generic Letter 87-09) were discussed. Based on its review to date, the staff noted that the proposal identifies each TS that would be changed by changing TS 3.0.4 to allow operational condition changes with inoperable equipment, provided the applicable action statements do not require a plant shutdown. However, safety analyses and determinations that there were no significant hazards considerations (NSHC) were not provided for each TS change. The staff stated that a safety analysis and NSHC determination should be made for each TS that was changed. Those TS which presently have an exception to TS 3.0.4 would not need to be analysed individually, provided deletion of the exception does not change the TS. In addition to safety analyses of plant specific changes to TS, assurance should be provided that the maintenance priorities will be such that plant startup will normally be initiated only when all equipment required to meet LCOs is operable. The staff scheduled October 21, 1988 as the target date for submitting additional information and scheduled an amendment issuance date of December 16, 1988.

The schedule dates for completion of other licensing actions is as follows:

October 30, 1988 - Extension of schedule for installation of RG 1.97 neutron flux monitors from RF03 to RF04.

January 15, 1989 - Increase reactor protection system surveillance intervals.

December 31, 1988 - Change fire protection license condition. This date is based on receipt of a response to Generic Letter 88-12 on October 30, 1988.

September 26, 1988 - Change TS to implement 10 CFR Part 55 regarding training of licensed personnel. This date was subsequently changed to October 28, 1988 per telephone call with M. Crawford (SERI) on September 30, 1988.

October 6, 1988 - Change fire protection zone because of room modifications. This amendment was issued September 23, 1988.

September 30, 1988 - Change position of PSRC member because of organization change. This amendment was issued September 21, 1988.

October 13, 1988 - Deletion of daily functional test in Rod Pattern Control System

October 21, 1988 - Add action statements for inoperable pressure and leakage instruments for the control rod scram accumulators.

The licensee stated a TS change would be submitted in October to replace position titles in Section 6.0 to preclude unnecessary TS changes when organization changes are made. A draft markup was provided and discussed (Enclosure 3). The licensee questioned whether the specific title of top management should remain. In an October 6, 1988 telephone call, the staff said that the Vice President, Nuclear Operations and the GGNS General Manager should remain as specific titles. Further, in response to the licensee's query, the staff said on October 6, 1988 that the ANSI/ANS Standard 3.1-1981 would not need to be referenced because the SRC member qualifications were stated in the TS, except for the qualifications of the SRC Chairman. Qualifications for the Chairman as stated in ANS 3.1-1981, Section 4.7.1b, should be included in that paragraph (See marked up TS pages 6-9, Enclosure 3).

The licensee stated that it will send a letter to withdraw its request of July 6, 1987 regarding isolation valve operability during shutdown.

Regarding followup information, the following comments were made:

- The documentation for completed license conditions will be provided by the licensee prior to RFO3.

- The procedures for use of the MSA-GMR-1 canisters to enter areas with a high airborne radioactivity were issued January 18, 1988 (OIS-08-04 Revision 12). Training lesson plans were revised January 19, 1988. Training will be a part of the regular training program; but if needed for use before training is completed, personnel will be trained prior to use.

- The licensee issued Material Noncomformance Report (MNCR) No. 0163-88 as followup to the NRC Information Notice (IN) 87-43 regarding gaps in the poison material found in Joseph Oats high density fuel racks at other facilities. Blackness tests were completed in the summer of 1988 and no gaps were found. Samples of the racks, both irradiated and unirradiated, indicated that some tearing or gaps may be present. Examinations and corrective actions, if needed, will be completed before new fuel arrives on site in January 1989.

- The licensee visually inspected springs in the main steam isolation valves (MSIV) in RFO1 as a followup to IN 86-81, which described broken springs found in Atwood Morrill MSIVs at Fermi. Inspections recommended by General Electric Company in SIL 422 were performed. The licensee talked to personnel at Detroit Edison Company who said the failures of the springs were plant specific. The NPRDS file showed that of 2704 total valves in 26 plants, the only failures of this type were the four valves at Fermi. SERI engineering department recommended tests of all replacements springs at 105% of the design load. Followup of IN 86-81, Supplement 1, will be done during a Project Manager site visit.

- The licensee said that the responses to Generic Letter 83-28 regarding the Salem ATWS event are applicable to both Units 1 and 2. For those responses which were submitted only on Unit 1, another letter will be sent to make responses applicable to Unit 2 also.

- The licensee will submit its analysis of the heat removal capability of the spent fuel pool cooling and cleanup system prior to RFO3. The present TS limit the number of fuel assemblies that can be put into the high density spent fuel racks until adequate cooling capacity is demonstrated. This limiting number of fuel assemblies in the present TS will not be reached until RFO5.

- The Project Manager advised the licensee of the San Onofre Unit 1 water hammer event of November 21, 1985, which was caused by failure of check valves. The valves failed because of repeated hammering of the disc stud and stud nut against the backstop during prolonged operation with low flow in the pipe. The licensee said it would study its systems for susceptibility to this type of failure. The Project Manager will obtain results of the study on a site visit.

The licensee stated that new TS submittals would be made as follows:

- September 1, 1988 - Deletion of leak rate tests for containment isolation valves in small diameter (1-inch) test, vent and drain lines.

- September 1, 1988 - Isolation of fuel transfer canal during refueling to work on transfer mechanism, if necessary.

- September 9, 1988 - TS changes related to the alternate decay heat removal system (ADHRS). The staff said that the design and safety analysis of the ADHRS should be submitted for the staff review and not just the changes to TS (new radiation monitors for plant service water and additional thermal overload protection devices for isolation valves).

- December 9, 1988 - Reload for RF03

The licensee's engineering personnel described the ADHRS design and operation (Enclosure 4). The system is designed to remove decay heat in the reactor starting 24 hours after shutdown. It would be used only during refueling outages, and then only a short time when both RHR trains are out of service. Design pressure is 250 psig, which is less than the design pressure of the LPCI-C piping to which it would discharge. The two pumps and two heat exchangers would be located in the RHR C Pump Room. Plant service water would be used in the heat exchangers.

/S/

Lester L. Kintner, Senior Project Manager
Project Directorate II-1
Divison of Reactor Projects I/II

Enclosure:
As stated

cc w/enclosure:
See next page

OFC	:LA:PD21:DRPR	:PM:PD21:DRPR	:D: DPRR/PD21	:	:	:	:
NAME	:P. Anderson	:L. Kintner	:E. Adensam	:	:	:	:
DATE	:10/17/88	:10/17/88	:10/18/88	:	:	:	:

OFFICIAL RECORD COPY

Mr. T. H. Cloninger
System Energy Resources, Inc.

Grand Gulf Nuclear Station (GGNS)

cc:

Robert B. McGehee, Esquire
Wise, Carter, Child, Steen and Caraway
P. O. Box 651
Jackson, Mississippi 39205

Nicholas S. Reynolds, Esquire
Bishop, Liberman, Cook, Purcell
and Reynolds
1400 L Street, N.W.
Washington, D.C. 20005-3502

Mr. Ralph T. Lally
Manager of Quality Assurance
Middle South Utilities System
Services, Inc.
630 Loyola Avenue, 3rd Floor
New Orleans, Louisiana 70113

Mr. John G. Cesare
Director, Nuclear Licensing
System Energy Resources, Inc.
P. O. Box 23054
Jackson, Mississippi 39205

Mr. C. B. Hogg, Project Manager
Bechtel Power Corporation
P. O. Box 2166
Houston, Texas 77252-2166

Mr. Ross C. Butcher
Senior Resident Inspector
U. S. Nuclear Regulatory Commission
Route 2, Box 399
Port Gibson, Mississippi 39150

Mr. William T. Cottle
GGNS Site Director
System Energy Resources, Inc.
P. O. Box 756
Port Gibson, Mississippi 39150

Mr. C. R. Hutchinson
GGNS General Manager
System Energy Resources, Inc.
Post Office Box 756
Port Gibson, Mississippi 39150

The Honorable William J. Guste, Jr.
Attorney General
Department of Justice
State of Louisiana
Baton Rouge, Louisiana 70804

Office of the Governor
State of Mississippi
Jackson, Mississippi 39201

Attorney General
Gartin Building
Jackson, Mississippi 39205

Mr. Jack McMillan, Director
Division of Solid Waste Management
Mississippi Department of Natural
Resources
Post Office Box 10385
Jackson, Mississippi 39209

Alton B. Cobb, M.D.
State Health Officer
State Board of Health
P.O. Box 1700
Jackson, Mississippi 39205

President
Claiborne County Board of Supervisors
Port Gibson, Mississippi 39150

Regional Administrator, Region II
U. S. Nuclear Regulatory Commission
101 Marietta Street
Suite 2900
Atlanta, Georgia 30323

ENCLOSURE 1

ATTENDEES

SERI-NRC MEETING AUGUST 31, 1988

NAME	AFFILIATION
M. L. Crawford	SERI
J. O. Fowler	SERI
J. K. Fortenberry	SERI
L. L. Kintner	NRC/PD2-1
J. G. Cesare	SERI
F. W. Titus	SERI
W. K. Hughey	SERI
R. J. Wright	SERI
A. Chu	NRC/PSB
M. Hartzman	NRC/MEB
M. W. Hodges	NRC/RXSB

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ENCLOSURE 2

AGENDA FOR 8/31/88 MEETING NRC - SERI

GRAND GULF NUCLEAR STATION, UNIT 1

REGARDING LICENSING ACTIVITIES

ROOM 14 B 9

8:30 a.m.	Review of TS changes Submittals, (NRC) DG 24 hour test, Snubber samples, Down travel cutoff of fuel hoist, TS 3.0 and 4.0, RG 1.97 flux monitor RPS Surveillance intervals., Deletion of Fire Protection (GL 88-12), 10 CFR 55 Training TS, Fire Detection TS, Deletion of daily functional Test in RPCS, new action statement for SCRAM accumulator TS, revised PSRC composition.
10:00 a.m.	New TS Submittals (SERI) - Remove position titles from TS 6.0 - Disposition of July 6, 1987 proposed change regarding isolation valve operability during shutdown
12:00 noon	LUNCH
1:00 p.m.	Followup Information (SERI) - Completion of OL Conditions - Status - Procedures for using MSA-GMR-1 canisters - IN 87-43 Gaps in high density spent fuel racks - IN 86-81 Broken springs in MSIVS - Applicability of GL 83-28 responses to Unit 2 - Additional heat removal capability for Spent fuel - Check valve reliability
2:00 p.m.	Alternate Decay Heat Removal System (SERI) Design Criteria, Function, Safety Analysis

DRAFT

6.0 AMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

Plant

6.1.1 The ~~GGNS General~~ Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Superintendent or, during his absence from the Control Room, a designated individual shall be responsible for the Control Room command function. A management directive to this effect signed by the ~~Vice President, Nuclear Operations~~ shall be reissued to all station personnel on an annual basis.

Senior Corporate Nuclear Officer

6.2 ORGANIZATION

6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for unit organization and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the UFSAR and updated at least annually.

Plant

b. The ~~GGNS General~~ Manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

Senior Corporate Nuclear Officer

c. The ~~Vice President, Nuclear Operations~~ shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 UNIT STAFF

The unit organization shall be subject to the following;

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2.2-1.

b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the reactor is in OPERATIONAL CONDITION 1, 2 or 3, at least one licensed Senior Reactor Operator shall be in the Control Room.

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UNIT STAFF (Continued)

c. A health physics technician* shall be onsite when fuel is in the reactor.

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

e. A site Fire Brigade of at least 5 members shall be maintained onsite at all times*. The Fire Brigade shall not include the Shift Superintendent, the STA, the two other members of the minimum shift crew necessary for safe shutdown of the unit, and any personnel required for other essential functions during a fire emergency. At least one AO shall be available to respond to non-fire-fighting commands from the Control Room.

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel.

Adequate shift coverage shall be maintained without routine heavy use of overtime. However, in the event that unforeseen problems require substantial amounts of overtime to be used, the following guidelines shall be followed:

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

2. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven-day period, all excluding shift turnover time.

3. A break of at least eight hours should be allowed between work periods, including shift turnover time.

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the ~~GGNS General~~ Plant Manager or his designee, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly

*The number of health physics technicians and Fire Brigade personnel may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.

(2)

UNIT STAFF (Continued)

Plant

by the ~~GGNS General~~ Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

g. The Operations Superintendent, Shift Superintendents, Operations Assistants, and Shift Supervisors shall each hold a Senior Reactor Operator License.

h. The Manager Plant Operations must have been a Senior Reactor Operator or have been certified on a plant of this type.

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine unit operating characteristics, NRC issuances, industry advisories, Licensee Event Reports, and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 The ISEG shall be composed of a multi-disciplined dedicated, onsite, group with a minimum assigned complement of five engineers or appropriate specialists.

INDEPENDENT SAFETY ENGINEERING GROUP (ISEG) (Continued)

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of unit activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

AUTHORITY

6.2.3.4 The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities or other means of improving unit safety to the ~~Vice President, Nuclear Operations~~ Senior Corporate Nuclear Officer.

6.2.4 SHIFT TECHNICAL ADVISOR

6.2.4.1 The Shift Technical Advisor shall provide technical support to the Shift Superintendent in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to safe operation of the unit.

6.3 UNIT STAFF QUALIFICATIONS

see attached insert 1

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions ~~and the supplemental requirements specified in Section A and C or Enclosure 1 of the March 28, 1980 NRC letter# to all licensees, except for the Chemistry/Radiation Control Superintendent who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975##, the Shift Technical Advisor who shall meet or exceed the qualifications referred to in Section 2.2.1.b of Enclosure I of the October 30, 1979 NRC letter to all operating nuclear power plants,~~ and those members of the Independent Safety Engineering Group used for meeting the minimum complement specified in Section 6.2.3.2, each of whom shall have a Bachelor of Science degree or be registered as a Professional Engineer and shall have at least two years experience in their field, at least one year of which experience shall be in the nuclear field.

Submitted in AECM-88/0146 dated August 3, 1988

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Training ~~Superintendent~~ Manager, shall meet or exceed INPO accreditation criteria ~~the requirements and recommendations of Section 5.5 of ANSI N18.1-1971~~ and ~~Appendix "A" of~~ 10 CFR Part 55, ~~and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter# to all licensees~~ and shall include familiarization with relevant industry operational experience.

Submitted in AECM-88/0146 dated August 3, 1988

6.5 Review AND AUDIT

6.5.1 PLANT SAFETY REVIEW COMMITTEE (PSRC)

FUNCTION

6.5.1.1 The PSRC shall function to advise the ~~GGNS General~~ Plant Manager on all matters related to nuclear safety.

*Not responsible for sign-off function.

~~#Except that the experience and other training information provided in the licensee's letter to the NRC dated July 29, 1985 are acceptable for the individuals listed in that letter.~~ (Deleted)

~~##Except that the individual identified in MP&L's letter to the NRC dated December 11, 1985 is considered qualified to hold the position of Chemistry/Radiation Control Superintendent based on the experience, education, and other information provided or referenced in that letter.~~

Submitted in AECM-88/0146 dated August 3, 1988

NL 88/10

Insert 1: ",except for the Radiation Protection Manager ~~Chemistry/Radiation Control Superintendent~~ and Shift Technical Advisor, who shall meet or exceed the education and experience requirements of ANSI/ANS 3.1-1981 as endorsed by Regulatory Guide 1.8, Revision 2, 1987, and licensed personnel who shall meet or exceed the criteria of the accredited license training program;"

COMPOSITION

6.5.1.2 The PSRC shall be ~~composed of the:~~ composed of 8 members who manage in the onsite organization at the Superintendent level or above.

~~Chairman:~~	~~Manager, Plant Support~~
~~Vice Chairman/Member:~~	~~Manager, Plant Operations~~
~~Member:~~	~~Manager, Plant Maintenance~~
~~Member:~~	~~Operations Superintendent~~
~~Member:~~	~~Technical Support Superintendent~~
~~Member:~~	~~Manager, Quality Services~~
~~Member:~~	~~Chemistry/Radiation Control Superintendent~~
~~Member:~~	~~I&C Superintendent~~
~~Member:~~	~~Plant Licensing Superintendent~~

(Deleted

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the ~~GGNS General~~ Plant Manager to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PSRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PSRC shall meet at least once per calendar month and as convened by the PSRC Chairman or Vice Chairman.

QUORUM

6.5.1.5 The quorum of the PSRC necessary for the performance of the PSRC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or Vice Chairman and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The PSRC shall be responsible for review of:

a. Station administrative procedures and changes thereto.

b. The safety evaluations for (1) procedures, (2) changes to procedures, equipment or systems, and (3) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question and all programs required by Specification 6.8 and changes thereto.

c. Proposed procedures and changes to procedures, equipment or systems which may involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed tests or experiments which may involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

e. Proposed changes to Technical Specifications or the Operating License.

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RESPONSIBILITIES (Continued)

f. Reports of violations of codes, regulations, orders, Technical Specifications, or Operating License requirements having nuclear safety significance or reports of abnormal degradation of systems designed to contain radioactive material.

g. Reports of significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

h. Review of all REPORTABLE EVENTS.

i. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

j. The plant Security Plan and changes thereto.

k. The Emergency Plan and changes thereto.

l. Items which may constitute a potential nuclear safety hazard as identified during review of facility operations.

m. Investigations or analyses of special subjects as requested by the Chairman of the Safety Review Committee.

n. Changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL, and radwaste systems.

AUTHORITY

6.5.1.7 The PSRC shall:

a. Recommend in writing to the ~~GGNS General~~ Plant Manager approval or disapproval of items considered under 6.5.1.6.a, c, d, e, j, and k, above.

b. Render determinations in writing to the ~~GGNS General~~ Plant Manager with regard to whether or not each item considered under 6.5.1.6.a, c and d, above, constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the SRC of disagreement between the PSRC and the ~~GGNS General~~ Plant Manager; however, the ~~GGNS General~~ Plant Manager shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The PSRC shall maintain written minutes of each PSRC meeting that, at a minimum, document the results of all PSRC activities performed under the responsibility and authority provisions of these Technical Specifications. Copies shall be provided to the SRC.

6.5.2 SAFETY REVIEW COMMITTEE (SRC)

FUNCTION

6.5.2.1 The SRC shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

who shall meet to qualifications in this section.

COMPOSITION

6.5.2.2 The SRC shall be composed of ~~the:~~ 10 members ~~who shall meet~~ or exceed the qualification requirements of Section 4.7 of ANSI/ANS 3.1-1981.

~~Chairman:~~	~~Vice President, Nuclear Operations~~
~~Member:~~	~~Vice President, Nuclear Engineering & Support~~
~~Member:~~	~~Director, Nuclear Plant Engineering~~
~~Member:~~	~~Site Director, GGNS~~
~~Member:~~	~~Director, Quality Programs~~ (Deleted)
~~Member:~~	~~Designated Representative, MSU System Services, Inc.~~
~~Member:~~	~~GGNS General Manager~~
~~Member:~~	~~Director, Nuclear Licensing~~
~~Member:~~	~~Manager, Radiological and Environmental Services~~
~~Member:~~	~~Manager, Operational Analysis~~

Two or more additional voting members shall be consultants to System Energy Resources, Inc. consistent with the recommendations of the Advisory Committee on Reactor Safeguards letter, Mark to Palladino dated October 20, 1981.

The SRC members shall hold a Bachelor's degree in an engineering or physical science field or equivalent experience and a minimum of five years of technical experience of which a minimum of three years shall be in one or more of the disciplines of 6.5.2.1a through h. In the aggregate, the membership of the committee shall provide specific practical experience in the majority of the disciplines of 6.5.2.1a through h.

add 4.7.1 b 6 years prof experience for chairman

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the SRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in SRC activities at any one time.

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AUDITS

6.5.2.8 Audits of unit activities shall be performed under the cognizance of the SRC. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Appendix A Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire unit staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. The Emergency Plan and implementing procedures at least once per 12 months.

f. The Security Plan and implementing procedures at least once per 12 months.

g. Any other area of unit operation considered appropriate by the SRC or the ~~Vice President, Nuclear Operations~~ Senior Corporate Nuclear Officer.

h. The Fire Protection Program and implementing procedures at least once per 24 months.

i. An independent fire protection and loss prevention inspection and audit shall be performed at least once per 12 months utilizing either qualified offsite licensee personnel or an outside fire protection firm.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 36 months.

k. The radiological environmental monitoring program and the results thereof at least once per 12 months.

l. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

m. The PROCESS CONTROL PROGRAM and implementing procedures for solidification of radioactive wastes at least once per 24 months.

n. The performance of activities required by the Quality Assurance Program to meet the criteria of Regulatory Guide 4.15, February 1979, at least once per 12 months.

(9)

AUTHORITY

Senior Corporate Nuclear Officer

6.5.2.9 The SRC shall report to and advise the ~~Vice President, Nuclear Operations~~ on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of SRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each SRC meeting shall be prepared, approved and forwarded to the ~~Vice President, Nuclear Operations~~ within 14 days following each meeting.

Senior Corporate Nuclear Officer

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the ~~Vice President, Nuclear Operations~~ within 14 days following completion of the review.

Senior Corporate Nuclear Officer

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the ~~Vice President, Nuclear Operations~~ and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.5.3 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.3.1 Activities which affect nuclear safety shall be conducted as follows:

a. Procedures required by Technical Specification 6.8 and other procedures which affect plant nuclear safety, and changes thereto, shall be prepared, reviewed and approved. Each such procedure or procedure change shall be reviewed by an individual/group other than the individual/group which prepared the procedure or procedure change, but who may be from the same organization as the individual/group which prepared the procedure or procedure change. Procedures other than Administrative Procedures shall be approved as delineated in writing by the ~~GGNS General~~ Manager. The ~~GGNS General~~ Manager shall approve administrative procedures, security implementing procedures and emergency plant implementing procedures. Temporary approval to procedures which clearly do not change the intent of the approved procedures may be made by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License. Temporary changes shall be reviewed by the reviewing authority within 14 days of being issued. For changes to procedures which may involve a change in intent of the approved procedures, the person authorized above to approve the procedure shall approve the change.

Plant

Plant

b. Proposed changes or modifications to plant nuclear safety-related structures, systems and components shall be reviewed as designated by the ~~Site Director, GGNS~~. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modifications. Implementation of proposed modifications to plant nuclear safety-related structures, systems and components shall be approved by the ~~GGNS General~~ Manager.

Senior Onsite Management Representative

Plant

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ACTIVITIES (Continued)

c. Proposed tests and experiments which affect plant nuclear safety and are not addressed in the Final Safety Analysis Report shall be reviewed by an individual/group other than the individual/group which prepared the proposed test or experiment.

d. Events reportable pursuant to Section 50.73 to 10 CFR Part 50 shall be investigated and a report prepared which evaluates the event and which provides recommendations to prevent recurrence. Such report shall be approved by the ~~GGNS General~~ Plant Manager.

e. Individuals responsible for reviews performed in accordance with 6.5.3.1.a, 6.5.3.1.b, 6.5.3.1.c and 6.5.3.1.d shall meet or exceed the qualification requirements of Section 4.4 of ANSI 18.1, 1971, as previously designated by the ~~Site Director, GGNS~~ Senior Onsite Management Representative or ~~GGNS General~~ Plant Manager, as applicable. Each such review shall include a determination of whether or not additional, cross-disciplinary review is necessary. If deemed necessary, such review shall be performed by the review personnel of the appropriate discipline.

f. Each review shall include a determination of whether or not an unreviewed safety question is involved.

g. Records of the above activities shall be provided to the ~~GGNS General~~ Plant Manager, PSRC and/or as necessary for required reviews.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified pursuant to the requirements of Section 50.72 to 10 CFR Part 50, and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PSRC and submitted to the SRC and the ~~Vice President, Nuclear Operations.~~ Senior Corporate Nuclear Officer

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The ~~Vice President, Nuclear Operations,~~ Senior Corporate Nuclear Officer and the SRC shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PSRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems or structures, and (3) corrective action taken to prevent recurrence.

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SAFETY LIMIT VIOLATION (Continued)

c. The Safety Limit Violation Report shall be submitted to the Commission, the SRC and the ~~Vice President, Nuclear Operations~~ Senior Corporate Nuclear Officer within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

i. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 4.15, February 1979.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed as required by 6.5, above, prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

6.8.3 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the:

1. RCIC system outside containment containing steam or water, except the drain line to the main condenser.
2. RHR system outside containment containing steam or water, except the line to the LRW system and headers that are isolated by manual valves.
3. HPCS system.
4. LPCS system.
5. Hydrogen analyzers of the combustible gas control system.

ALTERNATE DECAY HEAT REMOVAL SYSTEM

- Introduction
- Design Objectives and Requirements
- System Description
- Operating Modes
- Design Reviews
- Conclusions/Questions

INTRODUCTION

- Technical Specification 3/4.4.9 Basis:
  - Refueling Residual Heat Removal
  - Alternate Decay Heat Removal Methods

- Previous Alternative Decay Heat Removal Methods:
  - Reactor Water Cleanup
  - Fuel Pool Cooling and Cleanup
  - CRD System

- Problem Areas
  - Limited Capacity
  - Operating Flexibility

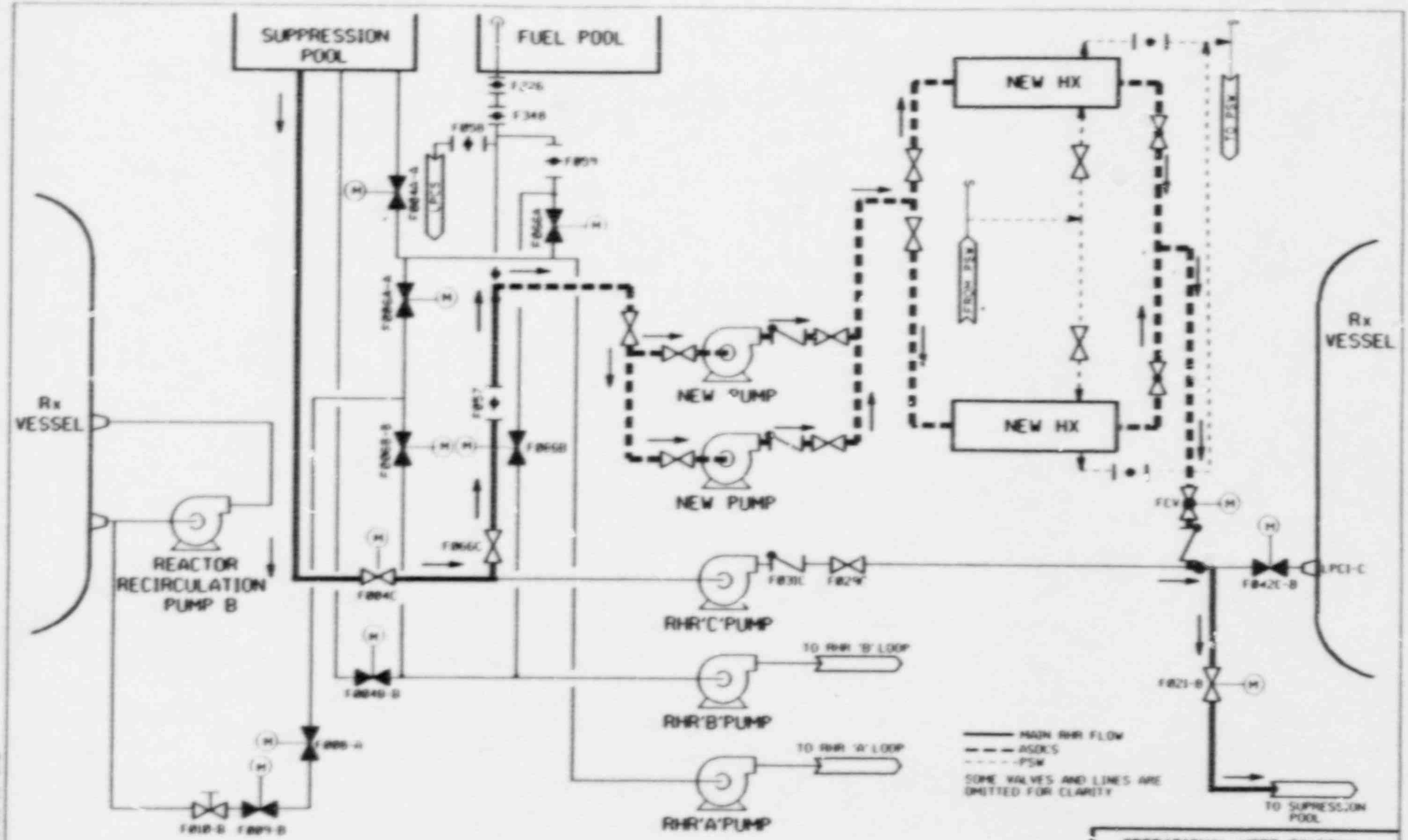
- Solution: New Plant System

ALTERNATE DECAY HEAT REMOVAL SYSTEM

- Design Objectives
  - Alternate Decay Heat Removal Capacity Available by the end of Outage Day 1
  - As Independent as Possible from Other Plant Systems
- Design Requirements
  - Maintaining Temperature Limits in Technical Specification Table 1.2
    1. 200F During Mode 4
    2. 140F During Mode 5
  - Operational in Modes 4 and 5 Only
  - No Safety Function Related To:
    1. Shutdown Capability
    2. Accident Mitigation
  - No Adverse Interaction with Existing Plant Systems
  - Pressure Boundary - ASME Section III Class 3, Seismic Category I
  - Operated from the Control Room

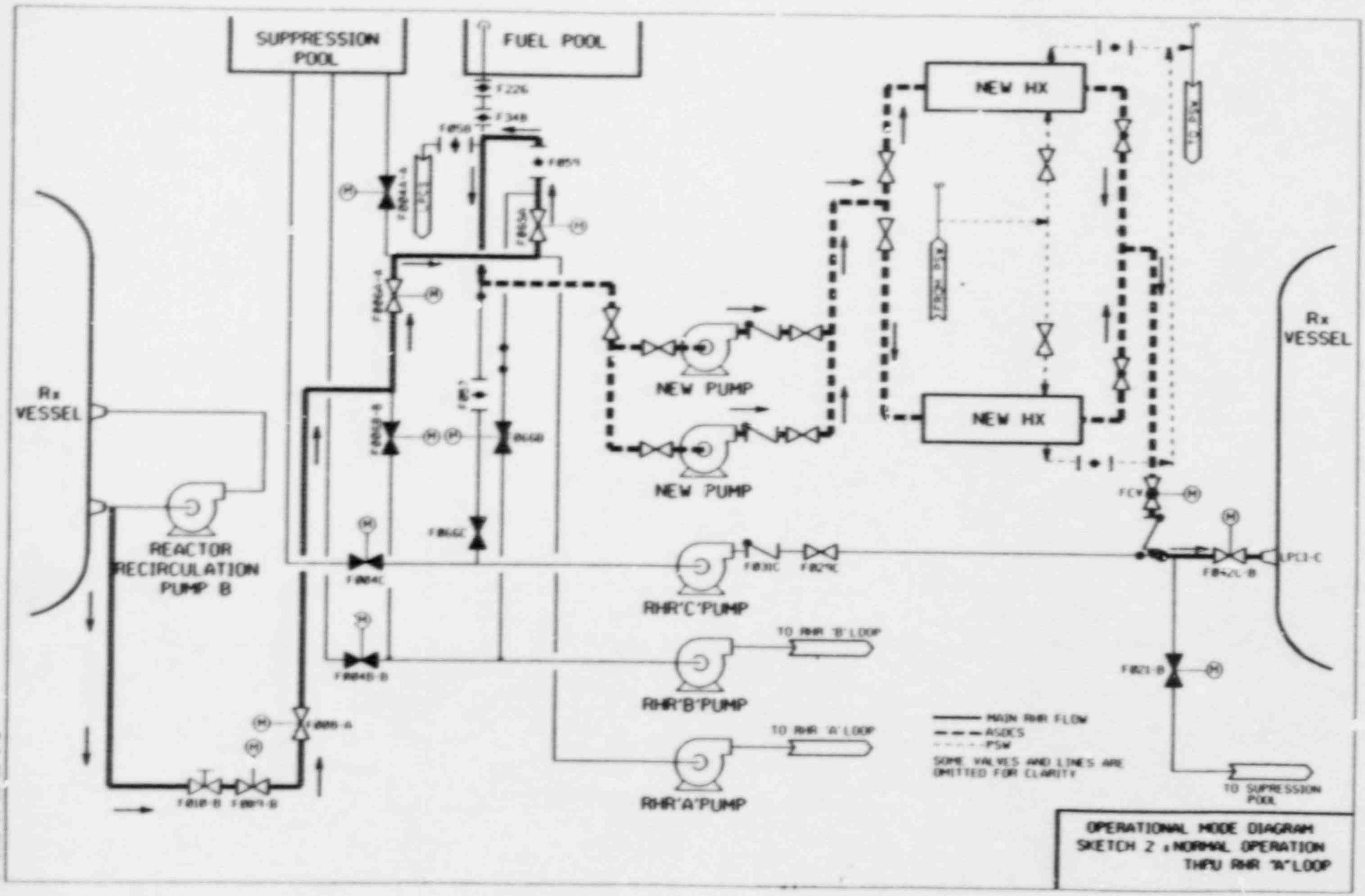
SYSTEM DESCRIPTION

- Primary Flowpath:
  - Suction Paths
    - RHR Common Suction
    - Spent Fuel Pool
  - Two Pumps and Heat Exchangers
  - Discharge Path: RHR "C"
  - System Interface
- Secondary Flowpath:
  - Plant Service Water System
  - Radiation Monitor on Effluent
- Electrical Power:
  - Non-Class IE
  - Exception: System Isolation Valve
- Control and Instrumentation:
  - Flow Control Valve
  - Temperature indication
  - Pump Controls and Indication
- HVAC:
  - Air Handling Unit
  - PSW Supplied

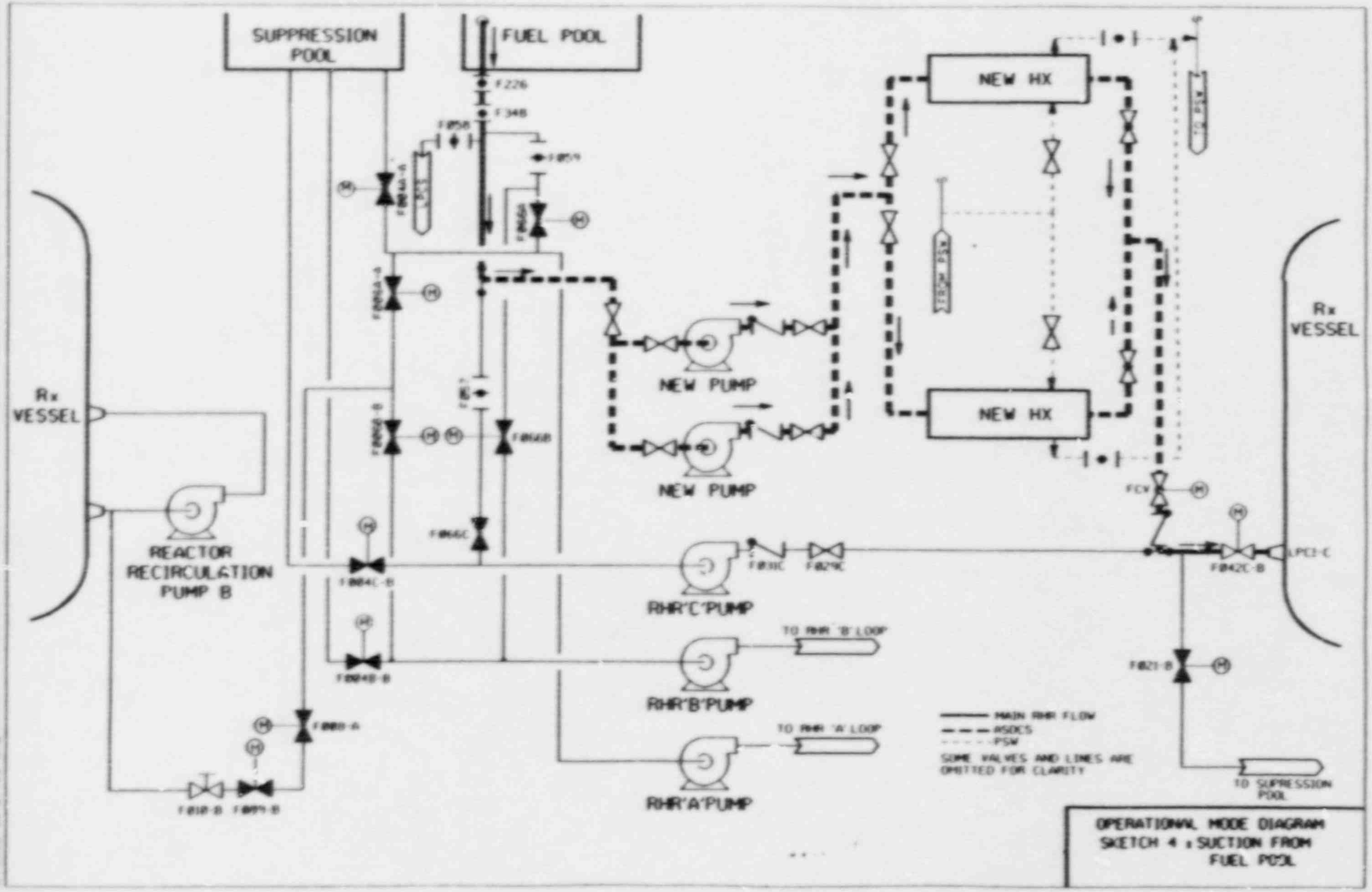


OPERATIONAL MODE DIAGRAM
SKETCH 1: TEST MODE SUCTION FROM SUPPRESSION POOL

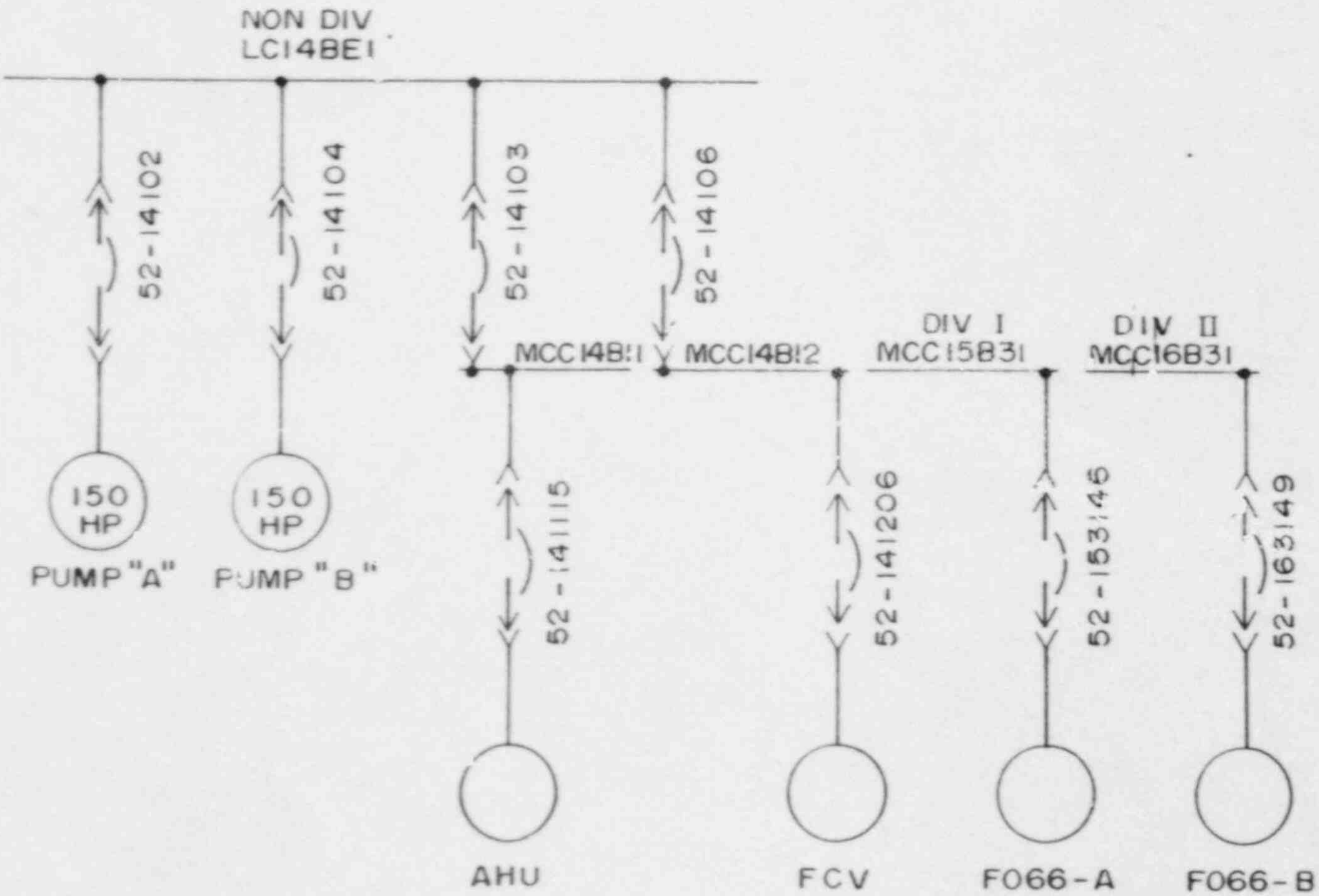
5


SUPPRESSION POOL
FUEL POOL
F226
F348
F050
F059
NEW HX
NEW HX
Rx VESSEL
Rx VESSEL
NEW PUMP
NEW PUMP
REACTOR RECIRCULATION PUMP B
F004C
F031C
F029C
RHR'C'PUMP
RHR'B'PUMP
RHR'A'PUMP
TO RHR 'B' LOOP
TO RHR 'A' LOOP
F066C
F067
FCV
LPCI-C
F042C-B
F021-B
F010-B
F009-B
TO SUPRESSION POOL
MAIN RHR FLOW
ASDCS
PSW
SOME VALVES AND LINES ARE OMITTED FOR CLARITY
OPERATIONAL MODE DIAGRAM
SKETCH 2 : NORMAL OPERATION
THRU RHR 'A' LOOP

6


SUPPRESSION POOL
FUEL POOL
F226
F348
NEW HX
NEW HX
NEW PUMP
NEW PUMP
Rx VESSEL
Rx VESSEL
REACTOR RECIRCULATION PUMP B
RHR'C'PUMP
RHR'B'PUMP
RHR'A'PUMP
TO RHR 'B' LOOP
TO RHR 'A' LOOP
FCV
LPCI-C
TO SUPRESSION POOL
MAIN RHR FLOW
ASDCS
PSW
SOME VALVES AND LINES ARE OMITTED FOR CLARITY
OPERATIONAL MODE DIAGRAM
SKETCH 4 : SUCTION FROM FUEL POOL

ADHRS I LINE



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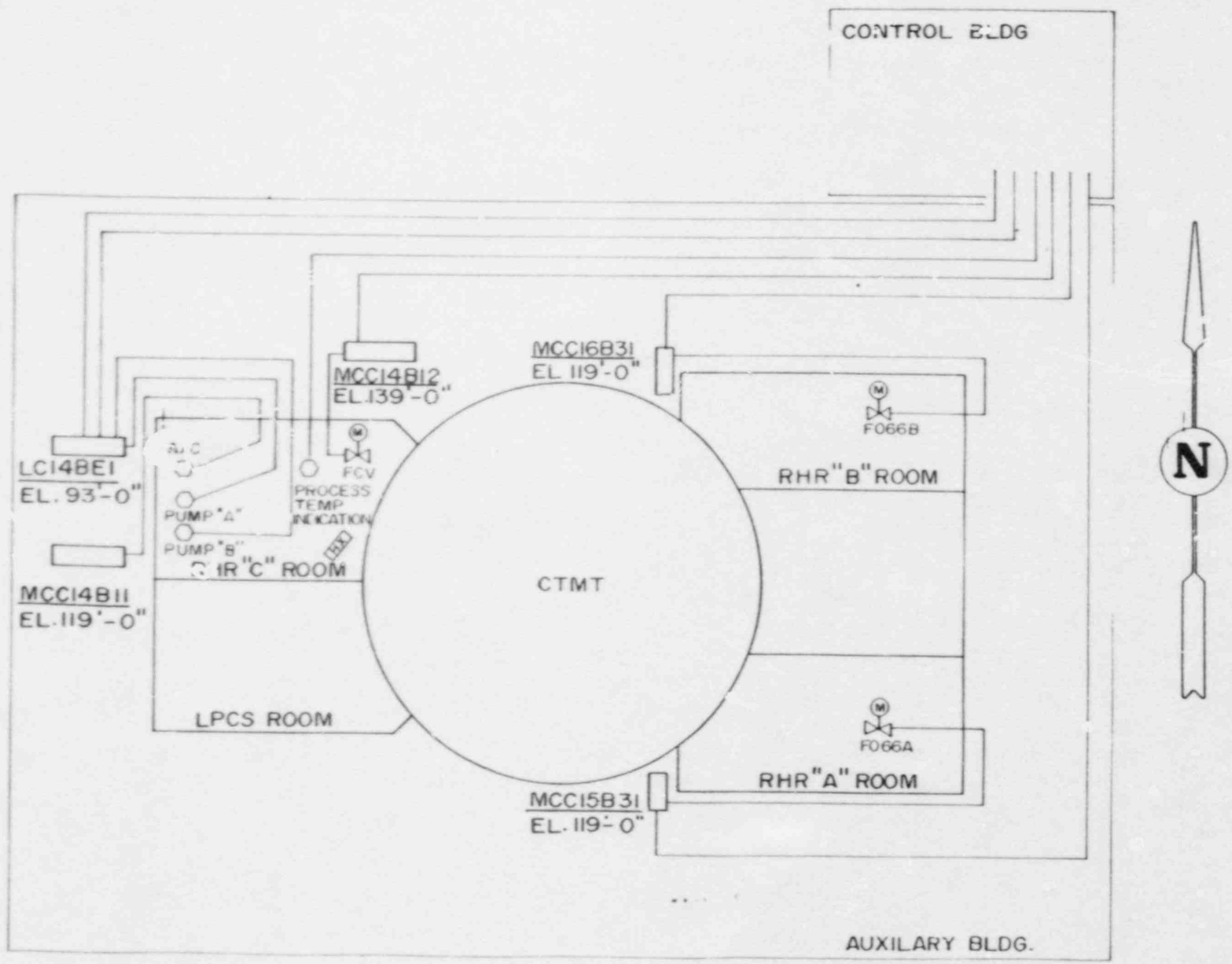
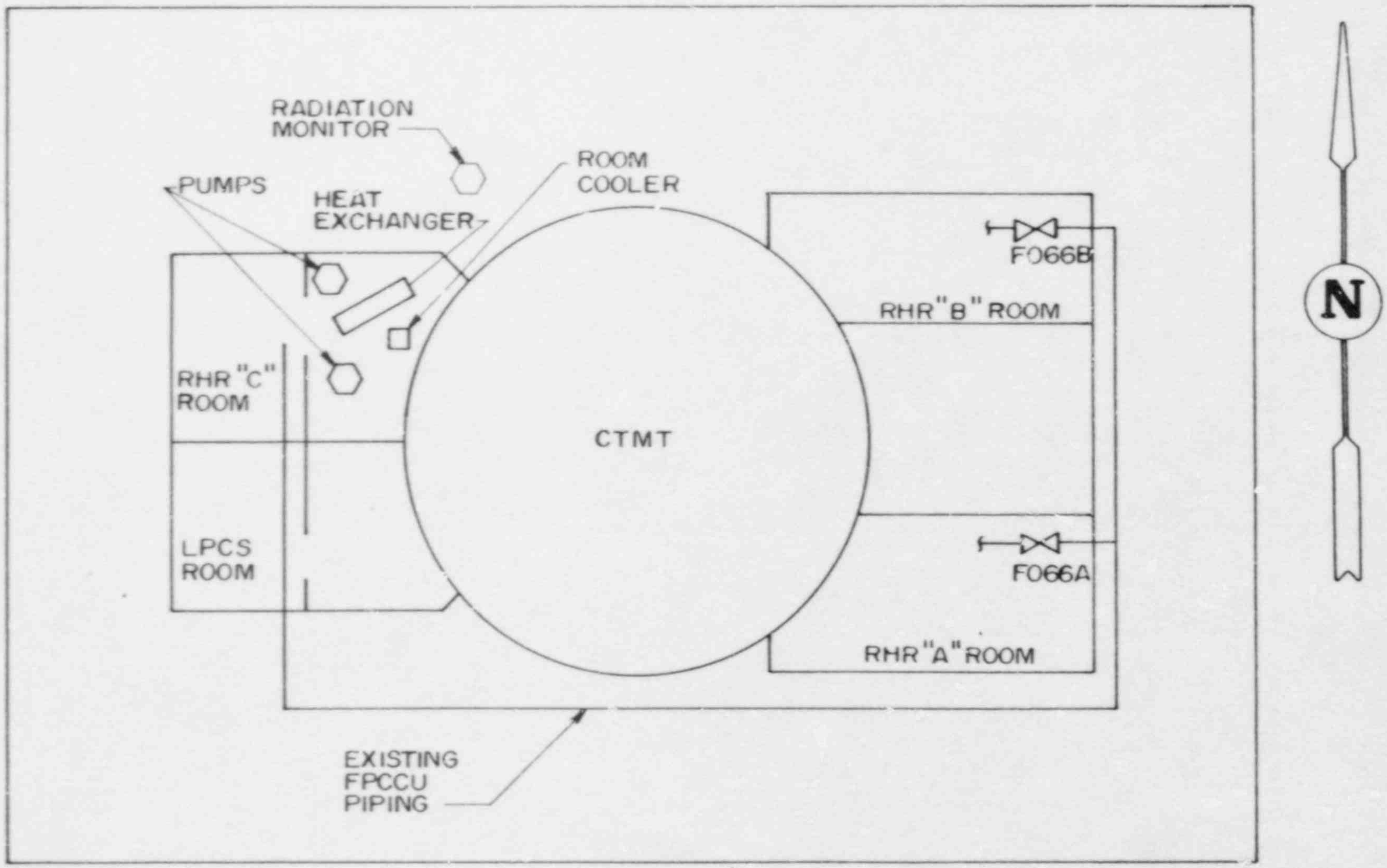


CONTROL BLDG
MCC16B31
EL. 119'-0"
MCC14B12
EL.139'-0"
LC14BE1
EL. 93'-0"
PUMP "A"
PUMP "B"
FCV
PROCESS
TEMP
INDICATION
RHR "C" ROOM
MCC14B11
EL.119'-0"
CTMT
F066B
RHR "B" ROOM
N
LPCS ROOM
F066A
RHR "A" ROOM
MCC15B31
EL. 119'-0"
AUXILARY BLDG.


N
RADIATION
MONITOR
ROOM
COOLER
PUMPS
HEAT
EXCHANGER
F066B
RHR "B" ROOM
RHR "C"
ROOM
CTMT
LPCS
ROOM
F066A
RHR "A" ROOM
EXISTING
FPCCU
PIPING

10

OPERATING MODES

- LAYUP
  - MODES 1, 2, 3
  - MECHANICALLY AND ELECTRICALLY ISOLATED FROM INTERFACING SYSTEMS
- FLUSH/TEST
  - SUPPRESSION POOL TO SUPPRESSION POOL
- RPV TO RPV COOLING
  - NORMAL COOLING LINEUP
  - COMMON SUCTION TO LPCI "C" FLOWPATH
- SPENT FUEL POOL TO RPV COOLING
  - USED WHEN "COMMON-SUCTION" LINE UNAVAILABLE

DESIGN REVIEWS AND SYSTEM INTERACTION EVALUATION

- Interfacing Systems
- Criteria
- Conducted in Two Parts
  - Functional Interactions
  - Physical Interactions

FUNCTIONAL INTERACTION EVALUATION

- Maintaining the Operability of Safety-Related Systems and Functions
- Operating Modes and Combinations of Operating Modes
- Potential for Inadvertent Drainage
- Controls and Operational Interactions

POTENTIAL FOR INADVERTENT DRAINAGE

- Evaluation Criteria
  - Single Active Component Failure
  - Single Operator Error
- Evaluation
  - Review Drain Paths Over 1 inch
  - Review Operating Mode Combinations
- Results
  - Feedback into Design Change
  - Clarified Administrative/Procedural Controls Requirements
- Conclusion
  - Alternate Decay Heat Removal System Presents no Greater Potential for Inadvertent Draining than Existing Systems

CONTROLS/OPERATIONAL INTERACTION EVALUATION

- Evaluation Criteria
  - Prevent Adverse Impact on Safety Related Systems
  - Assume Single Active Failure
  - Single Operator Error

- Evaluation.
  - Identify Potential Functional Interfaces
  - Consider Operating Mode Combinations
  - Consider Design Basis Events/Accidents
  - Developed Control Interaction Matrix

- Conclusion
  - Alternate Decay Heat Removal System Introduces no Adverse Control/Operational Interactions

PHYSICAL INTERACTIONS EVALUATION

- Effects on Plant Ambient Conditions (temperature, chemistry, etc.)
- Imposed loadings (normal, transient, seismic, etc. loads)
- Hazards Conditions (line breaks, flooding/spray, missiles, fire, etc.)
- Process Conditions (effects on reactor water flow, temperature, pressure, etc.)
- Radiological Effects

DISTRIBUTION FOR MEETING SUMMARY DATED: October 18, 1988

Facility: Grand Gulf Nuclear Station

Docket File
NRC PDR
Local PDR
PDII-1 Reading
E. Adensam
P. Anderson
A. Chu (8D1)
M. Hartzman (9H3)
W. Hodgcs (8E23)
OGC
E. Jordan (MNBB 3302)
B. Grimes (9A2)
ACRS (10)
B. Troskoski (17D19)

cc: Licensee/Applicant Service List

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