SYSTEM ENERGY RESOURCES, INC.

OLIVER D. KINGSLEY, JR.
Vice President
Nuclear Operations

September 23, 1988

U. S. Nuclear Regulatory Commission
Mail Station P1-137
Washington, D. C. 20555

Attention: Document Control Desk

Gentlemen:

SUBJECT: Grand Gulf Nuclear Station
Unit 1
Docket No. 50-416
License No. NPF-29
Alternate Decay Heat Removal System
Proposed Amendment to the Operating
License (PCOL-88/17)
AECM-88/0186

System Energy Resources, Inc. (SERI) is submitting by this letter a proposed amendment to the Grand Gulf Operating License. This amendment requests changes to the Grand Gulf Unit One Technical Specifications due to the proposed addition of the Alternate Decay Heat Removal System (ADHRS).

SERI has previously discussed the reasons for installing the new ADHRS with the NRC. The resulting improvement in alternate decay heat removal capability, the reduction in outage scheduling complexity and the intended use of the ADHRS were presented in a meeting between SERI and the NRC on August 15, 1988. In addition, on August 31, 1988 a meeting between SERI and the NRC was held to review the ADHRS design criteria, system function and system interaction analysis. A 10CFR50.59 evaluation on the ADHRS has been performed by engineering as part of the design process. Even though no unreviewed safety question was identified, SERI has elected to seek NRC review of the system as a whole including the Technical Specification changes that result from the system. Based on discussions with NRC staff on this subject, we believe that an NRC review under 10CFR50.90 is the most expeditious and efficient approach to license the use of the ADHRS.

The addition of the ADHRS is required in order to support the upcoming third refueling outage (RFO3) at Grand Gulf. As now scheduled, the third refueling outage is to begin approximately March 1, 1989. The RFO3 projected outage duration is 45 days or less. In order to support current outage schedules, SERI requests that the NRC complete its review of the ADHRS and the associated Technical Specification changes by no later than March 1, 1989. SERI is committed to assisting the NRC staff in completing its review.

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In accordance with the provisions of 10 CFR 50.4, the signed original of the requested amendment is enclosed and the appropriate copies will be distributed. The attachment provides the technical justification and discussion to support the requested amendment. This amendment has been reviewed and accepted by the Plant Safety Review Committee and the Safety Review Committee.

Based on the guidelines presented in 10 CFR 50.92, SERI has concluded that this proposed amendment involves no significant hazards considerations.

In accordance with the requirements of 10 CFR 170.21, an application fee of $150 is attached to this letter.

As noted, this amendment change is needed by March 1, 1989 to support the third refueling outage.

Yours truly,

for ODK

ODK:aly

Attachments:
1. Remittance of $150 Application Fee
2. Affirmation per 10 CFR 50.30
3. GGNS PCOL-88/17

cc: Mr. T. H. Cloninger (w/a)
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BEFORE THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

LICENSE NO. NPF-29

DOCKET NO. 50-416

IN THE MATTER OF

MISSISSIPPI POWER & LIGHT COMPANY
and
SYSTEM ENERGY RESOURCES, INC.
and
SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

AFFIRMATION

I, T. H. Cloninger, being duly sworn, state that I am Vice President, Nuclear Engineering & Support of System Energy Resources, Inc.; that on behalf of System Energy Resources, Inc., and South Mississippi Electric Power Association I am authorized by System Energy Resources, Inc. to sign and file with the Nuclear Regulatory Commission, this application for amendment of the Operating License of the Grand Gulf Nuclear Station; that I signed this application as Vice President, Nuclear Engineering & Support of System Energy Resources, Inc.; and that the statements made and the matters set forth therein are true and correct to the best of my knowledge, information and belief.

T. H. Cloninger

STATE OF MISSISSIPPI
COUNTY OF HINDS

SUBSCRIBED AND SWORN TO before me, a Notary Public, in and for the County and State above named, this 23rd day of September, 1988.

(SEAL)

Linda W. Miller
Notary Public

My commission expires:
My Commission Expires Aug. 5, 1992

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COVER PAGE

for

NL 88-12, Alternate Decay Heat Removal System

A. SUBJECT

I. NL 88-12 Alternate Decay Heat Removal System

II. Affected Technical Specifications:

- Table 3.3.7.1-1 (pages 3/4 3-59, 60), Radiation Monitoring Instrumentation

- Table 4.3.7.1-1 (page 3/4 3-62), Radiation Monitoring Instrumentation Surveillance Requirements

- Table 3.8.4.2-1 (page 3/4 8-48), Motor Operated Valves Thermal Overload Protection

B. DISCUSSION

I. INTRODUCTION

During refueling outages, the Residual Heat Removal (RHR) System shutdown cooling mode of operation is normally used to provide core cooling at the Grand Gulf Nuclear Station (GGNS). In this mode of operation, reactor coolant is pumped from the 'B' recirculation loop through a common suction line to either RHR A or B pumps and then on to the respective RHR heat exchanger to be cooled by the Standby Service Water System. The reactor coolant is returned to the vessel via either 'A' or 'B' feedwater lines depending on which RHR shutdown cooling loop is being used. The operation of RHR shutdown cooling during operational conditions 4 and 5 is controlled by Technical Specifications 3.4.9.2, 3.9.11.1 or 3.9.11.2.

In accordance with the present GGNS Technical Specifications, an ALTERNATE method of decay heat removal must be demonstrated operable for each inoperable loop of RHR shutdown cooling. In previous refueling outages, Reactor Water Cleanup (RWCU), Control Rod Drive (CRD), Fuel Pool Cooling and Cleanup (FPCCU) or other systems were used in various combinations for alternate decay heat removal. Because of the relatively limited decay heat removal capability of these systems, their use as alternate decay heat removal systems is restricted to time periods when decay heat loads are substantially reduced. For the upcoming GGNS refueling outage (RFO3) it is estimated, based on decay heat load data collected during previous refueling outages, that adequate alternate decay heat removal capacity using combinations of RWCU, CRD and FPCCU will not be available until approximately day 26 of the outage.

During each refueling outage one or both loops of the RHR shutdown cooling systems must be removed from service in order to perform required surveillances and/or routine maintenance. For example, the performance of local leak rate tests (LLRT's) on the RHR shutdown cooling common suction line requires that both loops of shutdown cooling be declared inoperable. In addition, required surveillances and/or maintenance activities on either Division I or

II RHR system components or supporting system components can require that either RHR shutdown cooling 'A' or 'B' loops be declared inoperable.

The third refueling outage (RFO3) at GGNS is now projected to begin approximately March 1, 1989. As currently scheduled, the RFO3 outage duration is expected to be approximately 45 days. The current schedule assumes additional alternate decay heat removal capability is available by day 13 of the outage. The RHR shutdown cooling common suction line work is currently scheduled to begin on day 14 of the outage. During this time, the reactor water level will be greater than 22 feet 8 inches above the top of the vessel flange. Thus in accordance with Technical Specification 3.9.11.1, one shutdown cooling loop of RHR is required to be operable and in operation. However; as previously noted, work on the common suction line results in both loops of RHR shutdown cooling being declared inoperable. In compliance with the action statement, one alternate decay heat removal method must be demonstrated operable. The present systems used for alternate decay heat removal (RWCU, CRD or FPCCU) are not estimated to have adequate heat removal capability until day 26 of the outage. Therefore, with the present alternate decay heat removal method, the earliest start date for the RHR common suction line work would be day 26 of the outage.

In addition to the common suction line work, the current RFO3 work scope has scheduled in-vessel vibration instrumentation removal. This work is now scheduled to begin on day 20 of the outage. In support of the vibration instrumentation work, the reactor cavity/upper containment pool water level must be lowered. During this period in accordance with Technical Specification 3.9.11.2, two shutdown cooling trains of RHR are required operable. The current schedule has planned during this period for RHR 'A' shutdown cooling to be operable. However, RHR 'B' shutdown cooling will be inoperable due to required Division II Emergency Core Cooling System (ECCS) testing and RHR 'B' local leak rate testing. Therefore in accordance with the action statements of 3.9.11.2, one alternate method of decay heat must be demonstrated. Again, due to limited alternate decay heat removal capacity, the earliest start date for the invessel vibration instrumentation removal work would be approximately day 35 of the outage.

As previously noted, the current RFO3 schedule has built into it the assumption that additional alternate decay heat removal capability will be available by day 13 of the outage thus allowing both RHR common suction line work and vibration instrumentation removal work to begin as scheduled. Without the added capability, these two work items would have to be rescheduled to later in the outage. The net effect of delaying this work to later in the outage is that the RFO3 critical path is extended by approximately 13 days.

It is the intent of System Energy Resources, Inc. (SERI) to conduct and manage all refueling outages in a safe and prudent manner. It is also SERI's goal to maximize plant availability by scheduling refueling outages such that the critical path is controlled by refueling activities and mandatory safety issues. As with the RFO3 schedule, it is estimated that future outages would be extended if additional alternate decay heat removal capability is not made available at GGNS. The impact on future outage duration is a result of delaying the lowering of the upper pool water level until approximately day 33 and thus impacts the vessel reassembly work. The net effect is that for future outages the refueling floor activities would be extended by approximately 13 days.

In order to eliminate the outage scheduling inflexibility due to limited alternate decay heat removal capacity and to ensure a possibility of a RFO3 duration of 45 days or less, System Energy Resources, Inc. (SERI) intends to add an Alternate Decay Heat Removal System (ADHRS) at GGNS. The ADHRS will use a combination of existing and new piping and valves to establish a flow path from either the reactor vessel 'B' recirculation loop through the common RHR shutdown cooling suction line or from the spent fuel pool. Reactor coolant will be pumped through the new ADHRS pumps and heat exchangers and back to the vessel via the RHR 'C' LPCI injection line. The new ADHRS equipment will be located in the RHR 'C' pump room. Functional control from the Control Room will be provided. Plant Service Water (PSW) will provide cooling water to the new ADHRS heat exchangers. ADHRS is designed to operate only in OPERATIONAL CONDITIONS 4 and 5 (see attached sketches for system alignments). During operational conditions 1, 2 and 3 the ADHRS will be isolated by locked closed or deenergized valves from the interfacing plant systems. The functional purpose of the ADHRS is not safety-related (i.e., NO ACCIDENT MITIGATION FUNCTION), however various portions of the ADHRS are safety-related in order to ensure that current plant safety-related requirements are not compromised by the installation or use of this system.

Two auxiliary features involved in the implementation of ADHRS will require changes to the GGNS Unit One Technical Specifications. These two features are the addition of the PSW radiation monitor and the addition of the motor operator/thermal overload devices to valves E12F066A and B. Justification and no significant hazards consideration for the ADHRS and its associated PSW radiation monitor and thermal overload devices are discussed in the following paragraphs.

As with the past refueling outages, it is the intent of SERI to minimize the time that both RHR shutdown cooling loops are unavailable for service and thus minimize the time that an alternate decay removal system other than RHR is required. Systems designated as alternate decay heat removal systems other than RHR will be demonstrated in actual tests to have adequate decay heat removal capability prior to the intentional removal of any RHR shutdown cooling loop from operability. This approach of identifying, analyzing and testing alternate decay heat removal paths is standard operating practice and has been successfully executed in past refueling outages at GGNS.

In addition to Technical Specification requirements, the intent of the SERI outage policy is to maintain at least one ECCS system and one Fuel Pool Cooling and Cleanup system functional at all times. At least one RHR pump and heat exchanger will be functional throughout the outage unless required maintenance or testing activities preclude this. Also, the diesel/generator associated with the above systems is required to be functional.

II. DESIGN APPROACH

A dedicated design team was assembled to develop the initial system objectives and design criteria and to evaluate the various design options. Representatives from SERI Nuclear Plant Engineering, SERI Project Management, Bechtel Power Corporation, and General Electric comprised the design team. The system design criteria was developed to ensure that appropriate functional requirements and safety considerations were incorporated into the evaluation of design options. The development of design options focused on reaching the goal of a 35 day outage duration.

To accomplish this goal, the following objectives were required to be addressed:

- Adequate alternate decay heat removal capability had to be provided by end of day 1 of the outage. This would ensure adequate decay heat removal capacity for all postulated conditions where it would be needed.

- A practical alternative to the RHR shutdown cooling common suction line was needed. This would provide an alternate suction path for those times when maintenance must be performed on the common suction line.

- The new design had to be as independent as possible from existing plant systems. The advantages gained from increased alternate decay heat removal capability would be diminished if too many restraints on other outage activities were created.

A matrix of options was developed, and a quantitative analysis of the design options was performed to obtain the optimum design goal.

The review of design options showed that an alternate suction path to the existing RHR shutdown cooling line was not practically achievable. This is due to the fact that no other line penetrating containment was available with the required flow capacity. However, an existing flowpath was found utilizing the common suction line and FPCCU piping that would meet the stated objective of an alternate suction path. No Unit 1 heat exchanger/pump combination was found that could provide the required cooling without imposing unacceptable restraints on outage activities. However, potential components were located in Unit 2 which is now under indefinite suspension. These two new heat exchanger/pump combinations were the Unit 2 FPCCU heat exchangers and pumps.

These components were evaluated and found to have adequate heat removal capability to maintain required reactor water temperatures at the end of day 1 of the outage. The location selected for installation of these components was the RHR C pump room. This area was selected because an existing suction path to this room via the FPCCU piping would minimize the amount of new piping required. In addition, ample space was available in the room for equipment installation. The LPCI 'C' discharge pipe was selected as a discharge path. A tie-in to the Plant Service Water System was selected as a cooling water source. This supply was chosen because of its close proximity to the RHR C pump room (which minimizes the amount of piping modifications), reliability of the PSW system, and demands on the PSW system are at a minimum during refueling outages.

Review of the conceptual design was performed by SERI Operations. With the addition of remote manual control of the new components, and reactor water temperature indication in the main Control Room, the concept was agreed upon. The SERI Plant Modifications and Construction Group performed walkdowns of the proposed design and verified the constructibility of the modification. Comments from these reviews were incorporated into the final design concept.

III. DESIGN FEATURES/HIGHLIGHTS

The following provides an overview of key ADHRS design features Additional detail on component design and features is provided in Section V.

The ADHRS uses a combination of existing and new piping and valves to accomplish the flow paths. The new equipment will be primarily located in the RHR 'C' pump room. Functional control is from the main Control Room. Heat removal is provided by the Plant Service Water (PSW) system. The functional purpose of the system is not safety-related (i.e., no accident mitigation function); however, various portions of the ADHRS are safety-related to ensure that current plant safety-related requirements are not compromised by the installation or use of the ADHRS.

Safety-related components include the reactor coolant side pressure boundary components (piping, valves, pumps and heat exchangers), added PSW piping and valves, and added motor operators and associated power supply and controls for valves E12F066A and B. A safety-related interlock is to be added for protection of the RHR 'C' pump from starting when its suction valve is closed during ADHRS operation. Additionally, a safety-related pump start permissive bypass switch is to be added to prevent a start of the RHR 'A' or 'B' pump when the associated E12F066A or B valve is open but a suction path does not exist. All components added by the ADHRS are supported to withstand safe shutdown earthquake (SSE) loads.

Major auxiliary equipment includes a nonsafety-related air handling unit for room cooling in the ADHRS area, and a nonsafety-related radiation monitor with an associated control room alarm for the PSW return from the ADHRS heat exchangers. These components are considered nonsafety-related in that they do not perform an accident mitigation function, are not required to function in order to shutdown the plant or do not serve as a reactor pressure boundary component. Portions of existing piping used in the ADHRS flow path will be shielded with lead wrap as necessary to maintain desired radiation zoning.

The ADHRS is designed to operate only in operational conditions 4 and 5. During operational conditions 1, 2 and 3 the ADHRS is isolated by locked closed or deenergized valves from connected plant systems.

Added components of the ADHRS are all located within the Auxiliary Building. With a design temperature of 200°F and a design discharge pressure of 250 psig, the ADHRS is designated as a moderate energy system. The system's designed maximum and minimum flowrates are approximately 3600 gpm and 1000 gpm respectively. The pressure boundary of the added process components and piping is primarily designated ASME III, Class 3. Piping and components employed for reactor coolant flow to and from the ADHRS is designated as ASME III, Class 2 or Class 3. PSW piping inside the RHR C pump room is ASME III Class 3, Seismic Category I up to the isolation valves at the existing supply and return headers. The ADHRS air conditioning unit's PSW piping is also ASME III Class 3, Seismic Category I up to the supply and return isolation valves. Other PSW air conditioning unit piping is ANSI B31.1 and designed for SSE loads. With the exception of the safety-related electrical components noted above, power supplies to the ADHRS components are from non-class 1E sources. This is considered consistent with the nonsafety-related function of ADHRS.

The ADHRS suction piping is connected to the RHR 'C' pump suction supply from the spent fuel pool, and the discharge piping is connected to the RHR 'C' loop discharge piping. Design operating modes of the ADHRS are the system flush/test mode, reactor pressure vessel to reactor pressure vessel cooling mode, and spent fuel pool to reactor pressure vessel cooling mode.

During operation, the ADHRS flow and cooling rate are controlled by a flow control valve operated from the main control room. System performance is monitored in the main control room by use of the existing RHR 'C' flow indication and added common heat exchanger inlet and outlet temperature indication, as well as existing reactor vessel temperature indication. System status is monitored in the main control room by position indication for valve E12F066A and B and the flow control valve, ADHRS pump running status lights, and position indication for various existing motor-operated valves in the ADHRS flow path.

IV. MODES OF OPERATION

The A^HRS is designed to be operated in the following modes:

a) Suppression Pool to Suppression Pool Flush/Test Mode

As shown in sketch 1, a suction path from the suppression pool through the E12F004C, E12F066C and G41F057 valves can be established to test ADHRS flow capabilities or to flush the system. The suppression pool water is then pumped through the ADHRS pumps, heat exchangers, flow control valve and back to the suppression pool via the RHR 'C' full flow test return line (E12F021). When operated in this mode, RHR C loop will be declared inoperable.

b) Vessel to Vessel Cooling Mode Via RHR A

Using the existing RHR shutdown cooling common suction line, reactor coolant is pumped through the E12F006A and E12F066A valves to the ADHRS (see sketch 2). Reactor coolant is cooled and returned to the reactor vessel through the ADHRS flow control valve and the RHR 'C' injection valve E12F042C.

c) Vessel to Vessel Cooling Mode Via RHR B

Using the existing RHR shutdown cooling common suction line, reactor coolant is pumped through the E12F006B and E12F066B valves to the ADHRS (see sketch 3). Reactor coolant is cooled and returned to the reactor vessel through the ADHRS flow control valve and the RHR 'C' injection valve E12F042C.

d) Spent Fuel Pool to Vessel Mode

A flow path may be established from the spent fuel pool via the G41F226 and G41F348 valves to the ADHRS heat exchangers as shown in sketch 4. As in the other modes, a return path through the LPCI 'C' piping is utilized. Operation in this mode is restricted to operational condition 5 when the upper cavity is flooded. During these times the reactor vessel may communicate with the Spent Fuel Pool via the fuel transfer tube. This flow path allows the RHR shutdown cooling common suction line maintenance to be performed with ADHRS providing the required cooling.

V. a) COMPONENT DESCRIPTION

A detailed discussion of the new components is provided below:

ADHRS Pumps - Two 50% capacity horizontal centrifugal pumps will be installed in the RHR C pump room which when operated in parallel will deliver approximately 3600 GPM at 265 ft TDH. The pumps are designed to the requirements of ASME Section III, Class 3. Adequate NPSH is provided for pump operation. Loads applied to the pump nozzles have been evaluated and are within the allowable value.

ADHRS Suction Piping - A tie in will be made to an existing line in the RHR C pump room to provide a flowpath to the new pumps. This piping will be ASME Section III, Class 3 and will be installed to Seismic Category I requirements. A manual gate valve is provided for isolation of the ADHRS from the FPCCU piping. In addition, manual gate valves are provided at each pump suction inlet for maintenance purposes.

ADHRS Pump Discharge Piping - Piping is provided for routing reactor water to the ADHRS heat exchangers. This piping is ASME Section III, Class 3 and is designed to Seismic Category I requirements. Piston type check valves and manual gate valves are provided at the discharge of each pump. In addition, manual gate valves are provided at the inlet to each heat exchanger.

ADHRS Heat Exchangers - Two 50% capacity u-tube heat exchangers will be used in a parallel arrangement to provide the required decay heat removal. The heat exchangers are designed to ASME Section III, Class 3 and Seismic Category I requirements. The heat exchangers will be installed in the RHR C pump room.

ADHRS Heat Exchanger Discharge - Piping is installed to allow reactor water to be routed from the heat exchangers to the LPCI 'C' injection line. This piping is ASME Section III, Class 3 up to the interface with the Class 2 portion of the RHR piping. At this junction, the new piping is Class 2. A swing check valve is provided in the Class 2 piping to provide isolation capability. A Class 2 globe valve is also provided at the interface. The primary function of the globe valve is for throttling of the ADHRS flow. However, this valve will provide additional isolation of ADHRS from RHR during operational conditions 1, 2, and 3. These components are designed to Seismic Category I requirements.

PSW Cooling Water Piping - Piping will be installed to provide PSW cooling water to the ADHRS heat exchangers. The tie-in to the PSW system will be made outside of the RHR C pump room to an existing PSW supply and return header located in the Auxiliary Building corridor at elevation 93'. This piping up to the isolation valves is designed to ANSI B31.1 requirements. PSW piping past the isolation valves which includes all piping inside the RHR C pump room is ASME III Class 3, Seismic Category I. The ADHRS air conditioning unit's PSW piping is also ASME III Class 3, Seismic Category I up to the supply and return isolation valves. Other PSW air conditioning unit piping is ANSI B31.1 and designed for SSE loads. Manual gate valves are provided outside of the RHR C pump room to isolate the ASME piping from the ANSI B31.1 piping. Also, manual gate valves and butterfly valves are provided at the heat exchanger inlets and outlets, respectively, to provide isolation capability.

Piping Materials - All piping is constructed of carbon steel. Stainless steel is used for instrument connections.

Lead Shielding - Lead wrap shielding will be installed on portions of existing FPCCU piping located in the Auxiliary Building corridor at elevation 93'. The shielding is being added so that existing radiation zone classification of these areas will be maintained.

FPCCU Isolation Valve - A manual ASME III Class 3 butterfly valve is being added to isolate the piping above elevation 122' from that portion of FPCCU piping that will be transporting higher activity reactor water. The isolation valve will effectively reduce the amount of lead shielding required.

Valve Motor Operators - Motor operators will be added to valves Q1E12F066A and B to allow the valves to be remote manually operated from the control room. These valves are normally locked closed and are used to align the RHR system in the spent fuel pool cooling assist mode. However, during operation of ADHRS the valves may be in the open position. The addition of the valve motor operators will ensure that the ASME Section III, Class 2 portion of the LPCI 'A' and 'B' suction piping can be isolated from the Class 3 FPCCU piping from the Control Room.

Air Conditioning Unit - A self-contained air conditioning unit will be installed in the RHR C pump room to maintain ambient room temperatures at normal levels during ADHRS operation. The air conditioning is non-safety related and is powered by a non-class 1E power source. The air conditioning unit is considered nonsafety-related in that it performs no accident mitigation function, is not required to function in order to shutdown the plant, or does not serve as a reactor pressure boundary component. Cooling water is supplied from the PSW system. Manual isolation valves are provided for the cooling water supply.

Radiation Monitor - A process radiation monitor is provided for the PSW cooling water discharge header. This non-safety related component will detect substantial intersystem leakage that could occur from potential heat exchanger tube failure and provides an alarm in the control room. The radiation monitor is considered nonsafety-related in that it performs no accident mitigation function, is not required to function in order to shutdown the plant or does not serve as a reactor pressure boundary component.

Miscellaneous Vents and Drains - High point vents and low point drains are provided as required to allow filling of the system prior to operation, and to allow for layup of the system during operational conditions 1, 2, and 3.

b) INSTRUMENTATION AND CONTROLS DESCRIPTION

Control of the ADHRS will be remote manual from the main control room. The instrumentation and controls provided for system operation are discussed below:

Pump Controls - Individual manual control of pump operation will be provided in the main Control Room. Pump running status lights will also display in the main Control Room. An automatic low suction pressure switch is provided to stop pump operation if the suction path is isolated, thereby protecting the pump from damage.

Process Indication - Flow indication is provided in the main Control Room by an existing flow element. Temperature sensors at the inlet and outlet of the ADHRS heat exchangers provide reactor water temperature indication in the main Control Room.

Flow Control Valve Control - Control of the ADHRS flow control valve is provided in the main Control Room.

Radiation Monitor Alarm - A high radiation alarm is provided in the main Control Room when abnormal radiation levels are detected in the PSW cooling water return piping. A low level alarm is provided to alert operators to potential radiation monitor failure.

Air Conditioning Unit Control - Local manual control of the air conditioning unit is provided.

Valve E12F066A/B Pump Start Permissive Bypass Switch - An existing pump start permissive will be bypassed during ADHRS operation. This bypass is needed to prevent the RHR A or B pumps from starting without a suction path. A control switch will be provided in the main control room to perform the bypass function.

RHR C Pump/Suction Valve Interlock - A safety-related interlock is being installed to prevent the RHR C pump from starting with the E12F004C valve closed.

VI. RADIATION MONITOR AND THERMAL OVERLOAD DEVICE

a) Radiation Monitor

(1) The ADHRS will utilize PSW to supply cooling water to the new ADHRS heat exchangers. A radiation monitor with associated control room alarms will be installed on the common PSW discharge from these new ADHRS heat exchangers. The purpose of this radiation monitor is to detect substantial intersystem leakage of reactor coolant into the PSW system.

(2) The radiation monitor being added to PSW will be identical in design to the two radiation monitors currently installed in the Standby Service Water (SSW) subsystems 'A' and 'B' as we l as the radiation monitor currently installed in the Component Cooling Water (CCW) system. Because these currently installed process radiation monitors are controlled by

Technical Specifications, SERI is requesting that this new PSW radiation monitor be added to the Technical Specifications.

b) Thermal Overload Device

(1) Valves E12F066A and B are currently manual valves and are normally locked closed. The current function of these two manual valves is to align RHR 'A' or 'B' loops in the spent fuel pool cooling assist mode as described in UFSAR Sections 5.4 and 9.1. These valves also provide a code boundary classification break between ASME Section III, Class 2 and Class 3 piping.

(2) In order to provide the ability for remote-manual alignment of RHR 'A' or 'B' during ADHRS operation, Class 1E motor operators are being added to these valves with handswitches located in the control room.

(3) Consistent with Regulatory Guide 1.106, thermal overload protection devices will be provided for these two Class 1E motor operators on valves E12F066A and E12F066B. Currently, thermal overload protection devices on Class 1E powered valves are controlled by Technical Specifications. As such, SERI is requesting that these new thermal overload devices be added to the Technical Specifications.

VII. PROPOSED TECHNICAL SPECIFICATION CHANGES

a) The new PSW radiation monitor will be added as Item 3 of Table 3.3.7.1-1 (pg. 3/4 3-59). Only one channel will be required operable anytime the new ADHRS heat exchangers are in operation. The applicable conditions of operability will be specified using a new footnote ## which will state:

"With ADHR system heat exchangers in operation."

No trip setpoint will be specified consistent with the alarm only function of this PSW radiation monitor. The alarm setpoint will be specified as less than or equal to 1 x 10E5 cpm. A measurement range of 10 to 10E6 cpm will also be specified. Action requirements in the event that the radiation monitor is inoperable will be provided through the use of ACTION 70. ACTION 70 will allow continued operation of the ADHRS with the PSW radiation monitor inoperable provided that a grab sample is obtained and analyzed at least once per 24 hours.

Surveillance requirements for the new PSW radiation monitor will be added as item 3 of Table 4.3.7.1-1 (pg. 3/4 3-62). The surveillance requirements will be identical to those of the CCW and the SSW radiation monitors. A channel check frequency of 'S' (i.e., at least once per 12 hours) will be specified. The channel functional test frequency will be 'M' (i.e., at least once per 31 days) and the channel calibration frequency will be 'A' (i.e., at least once per 366 days). The operational conditions for which these surveillances will be required will be consistent with the applicable conditions of operability. Again, this applicability will be specified using the same footnote defined above, but designated with the # character.

b) The new thermal overload protection devices for valves E12F066A and E12F066B will be added to Table 3.8.4.2-1 (pg. 3/4 8-48) and included with the other RHR system thermal overload devices. The bypass device column for these two new valves will read "Continuous", meaning that these thermal overload protection devices are continuously bypassed except during testing of the motor operator. The affected system column will read "RHR System".

VIII. PREVIOUS DISCUSSIONS AND TECHNICAL SPECIFICATION IMPROVEMENT PROGRAM

a) SERI has previously discussed the reasons for installing the new ADHRS with the NRC. The resulting improvement in alternate decay heat removal capability, the reduction in outage scheduling complexity and the intended use of the ADHRS were presented in a meeting between SERI and the NRC on August 15, 1988. In addition, on August 31, 1988 a meeting between SERI and the NRC was held to review the ADHRS design criteria, system function and system interaction analysis.

b) The Technical Specification changes proposed as a result of the new ADHRS are additional requirements to current Technical Specifications. SERI has evaluated these proposed changes using the criteria on addition/inclusion of items to the Technical Specifications provided by the Commission's Interim Technical Specification Policy Statement on Technical Specification Improvement and concludes that these proposed changes would not be included in the Technical Specification following implementation of this policy. As such, SERI requests that these proposed changes be examined using NRC guidelines for the implementation of the Interim Policy Statement on Technical Specifications Improvement to determine if the proposed changes should be added to the GGNS Technical Specifications.

C. JUSTIFICATION

I. SYSTEM DESIGN CRITERIA

The ADHRS is designed, fabricated, and installed to standards commensurate with its function and its interfaces with existing plant systems. The installation and use of the ADHRS has been evaluated to ensure that no adverse interactions with existing plant systems or safety functions will occur.

All components added by the ADHRS are supported to withstand safe shutdown earthquake (SSE) loads. Pressure boundary components are primarily designed and installed to ASME B&PV Code Section III, Class 3. This includes all piping and components handling reactor coolant (except at the Class 2 connection to the RHR C loop), and PSW piping from the ADHRS isolation valves inward (except at the air handling unit). Other piping is designed and installed to ANSI B31.1 criteria. Electrical power supply for the added motor operators for valves E12F066A and B is Class 1E, and is non-1E to other ADHRS components.

Safety-related components include all pressure boundary components (including supports) designated ASME Section III, Class 2 or 3, motor operators for valves E12F066A and B (including associated cable, conduit, and controls), the interlock for the RHR 'C' pump and suction valve, and the bypass switches on the RHR 'A' and 'B' pump start circuits. All other components added by ADHRS are non-safety grade.

Electrical installations are in accordance with GGNS commitments to Regulatory Guide 1.75. Detection of reactor coolant intersystem leakage, in conformance with Regulatory Guide 1.45, is provided for by a radiation monitor at the common PSW return from ADHRS. The design and installation maintains fire protection commitments to 10CFR50 Appendix R. The ADHRS quality classifications and seismic design meets or exceeds the guidelines of Regulatory Guides 1.26 and 1.29 for systems containing radioactive materials. The ADHRS cooling function is not safety-related, and this function is not credited for the satisfaction of any General Design Criteria.

The ADHRS design and operation has been evaluated for interactions with existing plant systems to ensure that adverse affects would not exist and that appropriate procedural controls and restraints are included as a part of the modification. System evaluation criteria included avoiding adverse effects on existing safety systems and functions, avoidance of adverse effects to safety-related aspects of the ADHRS, and avoidance of unacceptable offsite effects. Complete details of the evaluation are contained in the attached Engineering Report (Attachment 1, ADHRS Interaction Evaluation). A brief summary is provided below:

o Interconnected system compatibility - assurance of the continued operability and functions of physically connected systems was made. These systems included RHR (for shutdown cooling, LPCI modes), fuel pool cooling and cleanup system (for cooling and shielding), LPCS (for LPCS) and nuclear boiler system (for temperature control, coolant circulation, shielding, and coolant inventory). Potential concurrent operation of the ADHRS in various modes, and RHR and LPCS in various modes was evaluated and unacceptable combinations prohibited.

o Inadvertent drainage evaluation - the potential for inadvertent drainage paths due to the ADHRS was evaluated and appropriate procedural controls and design changes identified.

- o Controls/operational interactions - the satisfactory fulfillment of safety-related functions for design basis events was assured. This included step-by-step reviews of various event and ADHRS operating mode combinations. Pertinent events included reactor high pressure, LOCA, and loss of shutdown cooling. For LOCA events, for example, the ability of the assumed operable ECCS trains to deliver flow to the reactor vessel in accordance with operational conditions 4 and 5 requirements was examined. Criteria included maintenance of keep-fill and minimum flow functions, maintenance of the ASME Class 2 boundary of the ECCS flow path, avoidance of excess flow or NPSH concerns, avoidance of overpressurization of the ADHRS system, etc. Single failures and operator errors were examined for their effects as well. Necessary procedural controls and design changes were identified.

- o Physical interactions - Physical interactions between the ADHRS and other plant systems, structures and components were evaluated for acceptability. These included:

    - Ambient conditions - temperature, radiation, materials and reactor water chemistry

    - Imposed loadings - by the ADHRS and on the ADHRS including seismic, hydrodynamic, waterhammer, thermal, fatigue, etc.

    - Hazards conditions - by the ADHRS and on the ADHRS including seismic II/I, pipe break, flooding, missiles, etc. consistent with UFSAR requirements.

    - Process conditions - flow, temperatures, and pressure

    - Offsite effects - due to a release of radioactive reactor coolant.

Various criteria associated with the above conditions were evaluated and found to be satisfied, including maintenance of environmental qualification of safety-related equipment (including addition of pipe shielding and a room cooler as part of the ADHRS modification), and structural integrity of the ADHRS and other plant systems. Offsite effects were examined by analyzing a postulated ADHRS heat exchanger tube rupture accident. Doses at the Site Boundary were found to be less than a small fraction of 10CFR Part 100 limits. The effects of an anticipated operational occurrence of a heat exchanger tube crack on drinking water concentration was also analyzed. The resultant radioactive concentrations at the nearest potable water supply on the Mississippi were found to be well below those allowed by 10CFR Part 20.

II. RADIATION MONITOR

a) The purpose of the new PSW radiation monitor is to detect substantial intersystem leakage of reactor coolant into the PSW system per guidance provided by Regulatory Guide 1.45. Because this monitor is not required to prevent safety limits (accidental offsite doses) from being exceeded, it is not designated as safety-related or powered from a Class 1E power source, and only one monitoring channel is provided. Also, the radiation monitor will perform no automatic isolation function. This is consistent with the design of radiation monitoring for the SSW and CCW systems. Like the SSW and CCW systems, a high radiation alarm is provided in the Control Room to alert operators to a condition that may require system isolation. A low radiation alarm is also provided to indicate potential monitor failure. In the event of a low radiation alarm, periodic grab samples could be taken to allow continued system operation as currently stipulated in the Technical Specification for the SSW and CCW radiation monitors.

An offsite radioactive release from the ADHRS would only occur due to a failure of the ADHRS pressure boundary. The ADHRS pressure boundary is designed to ASME Section III, Class 3 and seismic category I requirements. A failure of this Seismic Category I boundary must be postulated in order to have a potential for off-site radioactive release. This highly unlikely event is considered to fall under the classification of postulated accidents in accordance with FSAR Section 15.0.3.

The potential for reactor coolant leakage to the environment was assessed in a conservative, bounding analysis which assumed that the entire reactor coolant volume at the Technical Specification reactor coolant activity limit (decayed for 24 hours) was released. (This assumption is equivalent to postulating the highly unlikely gross structural failure of the ASME III, Class 3, Seismic Category I ADHRS heat exchanger pressure boundaries.) The analysis further assumed that 10 percent of the iodine in the reactor coolant was immediately transported, unfiltered, to the site boundary using the 0-2 hour atmospheric dispersion factor.

The resultant offsite boundary doses are 1.19 rem thyroid and 0.00137 rem whole body.

These doses are significantly less than a small fraction of the 10CFR100 limits applicable for this event (10% x 300 rem = 30 rem thyroid; 10% x 25 rem = 2.5 rem whole body).

The GGNS UFSAR Section 15.7.2 addresses the consequences of a postulated unexpected and uncontrolled release of radioactivity due to a radioactive liquid waste system failure. The results of the assessment provided in Table 15.7-7 are a 1.25 rem thyroid dose and a "negligible" whole body dose. Considering a whole body dose of 0.00137 rem to be negligible, this assessment bounds the ADHRS heat exchanger rupture.

The limits in Regulatory Guide 1.26 Rev. 3 and 1.29 Rev. 3 for non-seismically designed equipment containing radioactive materials are 0.5 rem whole body or the equivalent to an organ, which is 3.0 rem for the thyroid. Although the ADHRS heat exchangers are seismically designed, the calculated doses from a postulated failure of this equipment is nevertheless below the guidelines for non-seismically designed equipment.

b) The ADHRS will be available as an alternate decay heat removal system only during operational conditions 4 and 5. During other operational conditions, the ADHRS is designed to be isolated by locked closed or deenergized valves from all other interfacing systems. Only when the ADHRS is operating or not mechanically isolated is there an interface between PSW and reactor coolant. Consistent with this configuration, the applicable conditions for operability of the proposed radiation monitor will be whenever the ADHRS is in operation or otherwise not isolated.

c) The nominal radiation high alarm setpoint is selected as less than or equal to 1 x 10E5 cpm. The current Technical Specification setpoint for the SSW and CCW detectors is also less than or equal to 1 x 10E5 cpm. This setpoint was evaluated and determined to be appropriate for the radiation monitor based on the following:

(1) The PSW radiation monitor is identical to the SSW and CCW monitors.

(2) The PSW radiation monitor will be located in an area of similar background radiation as compared to the SSW and CCW monitors.

(3) PSW, SSW and CCW radiation monitors perform similar tasks of monitoring for potential intersystem leakage of reactor coolant into cooling water.

(4) The potential cooling water radionuclide concentration levels would be similar to the SSW and CCW systems.

d) The specified measurement range for the SSW and CCW radiation monitor is 10 to 10E6 cpm (Table 3.3.7.1-1). The PSW radiation monitor is identical to both the SSW and CCW monitors. Therefore, the proposed specified measurement range for the PSW radiation monitor of 10 to 10E6 cpm is considered appropriate.

e) The required ACTION being proposed in the event that the PSW radiation monitor is inoperable allows continued operation provided that grab samples of the monitored parameter are obtained and analyzed at least once per 24 hours. Again, this action requirement is being proposed consistent with the action requirements for the SSW and CCW radiation monitors. As previously discussed, the dose consequences of a case in which the entire reactor coolant inventory is released were determined to be significantly less than regulatory guideline limits. The proposed 24 hour grab sample interval is bounded by the calculated offsite doses resulting from a postulated gross structural failure of the ADHRS pressure boundary.

f) The surveillance requirements being added for the PSW radiation monitor are identical to the current surveillance requirements for both the CCW and the SSW radiation monitors. The PSW radiation monitor being added is identical to the CCW and SSW radiation monitor. Therefore, the surveillance requirements of these monitors are, appropriately, identical.

III. THERMAL OVERLOAD PROTECTION DEVICE

a) The purpose of the motor operated valve thermal overload protection device is to protect the motor operator under overload conditions. In order to ensure that safety-related motor operated valves, whose motors are equipped with thermal overload protection devices, will perform their function, Regulatory Guide 1.106 (Rev. 1, March 1977) provides two alternatives: (1) bypassing the thermal overload protection device, either continuously or only under accident conditions and (2) establishing a trip setpoint of the thermal overload protection device with all uncertainties resolved in favor of completing the safety-related function.

b) Currently, valves E12F066A and B are manually operated valves. During normal operation, these valves are locked closed. The valves are opened to establish either RHR 'A' and/or 'B' loop in the spent fuel pool cooling assist mode of operation as indicated in UFSAR Sections 5.4 and 9.1 and also for pump testing. Automatic or remote-manual actuation of these valves is not required in order to mitigate the consequences of an accident. These valves serve as the code boundary classifications break between ASME Section III, Class 2 and Class 3 RHR piping. The failure of the valve actuators will have no functional effect on ECCS (i.e., LPCI 'A', 'B', or

'C') system operation. The failure of E12F066A or B to close or to be closed by the operator prevents the establishment of an ASME Section III Class 2 pressure boundary for ECCS as recommended in Regulatory Guide 1.29 and committed to in UFSAR Section 3.2. However, as previously noted this failure has no functional effect on the operation of ECCS. The ECCS system has been evaluated with these valves failed in the open position and was found to be capable of delivering the required ECCS system flow rates. In addition, the piping beyond the E12F066A and B valves is safety-related (Class 3), Seismic Category I and designed for the process conditions that would occur in the LPCI mode.

The addition of the Class 1E, AC motor operators to E12F066A and B will allow for Control Room alignment of ADHRS during operational conditions 4 and 5. During operational conditions 1, 2 and 3 these valves will be closed. The motor operators also allow remote manual isolation from the Control Room for code boundary restoration of ECCS piping during periods when ADHRS is in operation.

c) The thermal overload protection devices on the motor operators being added to valves E12F066A and B will be bypassed continuously. The thermal overload devices will only be placed in service when ever the valve motor operator is undergoing periodic testing. This will ensure that the safety-related motor operated valve will perform its intended function when required. As noted above, the failure of E12F066A and B would however have no functional effect on ECCS with the exception that the required coded boundary would not be established. In order to ensure the ECCS code boundary can be restored from the Control Room, with a high degree of assurance, it is proposed that the thermal overload devices be bypassed continuously as directed by Regulatory Guide 1.106 (Rev. 1, March 1977).

D. NO SIGNIFICANT HAZARDS CONSIDERATIONS

The following analysis about the issue of no significant hazards consideration, using the standards of 10CFR50.92, is provided in accordance with 10CFR50.91(a).

I. These changes would not significantly increase the probability or consequences of an accident previously evaluated.

a) The ADHRS (including operating requirements) has been found not to significantly increase the probability of previously evaluated accidents and events as discussed below.

The nature of the ADHRS and its interconnections with existing plant systems results in specific consideration of the following events or possible increases in probability:

- o Pipe break or leak
- o Loss of coolant accident (LOCA)/inadvertent reactor vessel drainage
- o Internally generated missiles
- o Reactor coolant pressure boundary (RCPB) overpressure
- o Fire
- o Loss of electrical power
- o Offsite radioactive release

Other events/accidents are excluded from detailed consideration because there are no identified functional (physical connections or controls interface) relationships and because the ADHRS has been designed to preclude any adverse spatial or environmental interactions. Examples of such events include secondary plant transients, loss of instrument air, reactivity control failures, and fuel handling accidents.

Pipe Break or Leak

The ADHRS will not significantly increase the probability of a pipe break or leak in existing systems, and the probability of failure of the ADHRS fluid components is no greater than existing systems, as presently analysed in UFSAR Section 3.6.

The design of the ADHRS is such that stress loadings due to the attachment of ADHRS piping and equipment to existing plant fluid systems does not cause applicable stress allowables on existing components to be exceeded. This is documented by calculations that employed methodologies, assumptions, loading combinations, and other criteria the same or equivalent to those contained in UFSAR Section 3.9.

Evaluated were mechanical loadings imposed by the ADHRS, changes in the design pressure and temperature of existing piping, and fatigue considerations. Attachment loadings were evaluated at the suction and discharge connections of the ADHRS to the RHR and PSW systems, at valves E12F066A and B due to the addition of motor operators to these valves, loads imposed by the new valve added to the vertical leg of the RHR suction line from the spent fuel pool, and at locations where lead wrap pipe shielding was added. Existing piping and supports were verified to be adequate or supports are to be modified and/or new supports added as applicable to achieve acceptable conditions.

The added ADHRS piping handling reactor coolant is designated as safety-relatec, ASME Section III, Class 2 and Class 3, and Seismic Category :. In addition, PSW piping inside the RHR 'C' pump room and up to the isolation valves at the existing supply and return headers and air handling unit is designated as safety-related, ASME Section III, Class 3, and Seismic Category I. Other piping is B31.1 and designed for SSE loads. The design and analysis of this piping is to the same standards, criteria, and methodology as existing plant piping. Damage to the ADHRS piping and components due to over-pressure is prevented by design of the system for the highest pressure possible from ADHRS operation (pump shutoff head plus maximum static head at the pump suction), and providing for system isolation from higher pressure sources (i.e., RHR 'C' pump discharge). The portion of the system that cannot be isolated from RHR 'C' (downstream of the ADHRS isolation valves) is designed for RHR 'C' design pressure. Thermal relief valves are provided in the ADHRS for overpressure protection when the system is isolated and for minor leakage that may occur across the boundary valves. Adequate vents and drains are included in the design to provide for complete filling and venting prior to pump start. This provision will preclude the possibility of a waterhammer event during startup of the system.

Materials (mostly carbon steel) used in the ADHRS pressure boundary are compatible with those used in connected systems and will not induce a degradation of existing piping. The ADHRS does not employ any stainless steel pressure boundary components other than instrument connections. Reactor coolant chemistry is controlled to prevent failure of stainless steel components due to intergranular stress corrosion cracking. On the reactor coclant side of the ADHRS, no features are provided that would alter reactor coolant chemistry.

Erosion of existing and new piping is not expected to occur based on flow velocities and expected usage times. Specified nominal corrosion allowances for new piping are the same as for existing piping, and excessive corrosion during lengthy inactive periods is to be precluded by wet layup of the reactor water side (the same as currently performed for the existing RHR system), and layup of the PSW side.

Adequate protection from overpressurization of piping is provided by the inclusion of appropriate isolation capability at system interfaces.

Consequently, the ADHRS modification does not significantly increase the probability of a break in existing piping and the probability of a break in the new piping is no higher than in existing piping.

LOCA/Inadvertent Vessel Drainage

Such events as a consequence of the presence or use of the ADHRS can only occur during operational conditions 4 or 5 (cold shutdown or refueling), since in all other operational conditions the ADHRS is required to be physically isolated from other systems.

The probability of a pipe break outside containment is not significantly increased by the addition of the ADHRS. Per UFSAR Section 3.6, postulating a pipe failure when the ADHRS is aligned or operating is not required. Specifically, UFSAR Section 3.6A.1.1.c states that "Pipe breaks or cracks were postulated to occur during normal plant operation (i.e., reactor startup, operation at power, hot standby or reactor cooldown to cold shutdown).". These conditions do not apply to operational conditions 4 or 5 when the ADHRS would be in operation.

To the extent that the criteria cited in Section 3.6 implicitly assume adequate piping design, the ADHRS satisfies this assumption. As for a pipe break inside containment, the only piping involved is the RHR 'C' LPCI injection line to the reactor vessel. This line was evaluated for an increased fatigue usage factor arising from a longer operating time for this pipe (albeit at a lower flow rate) due to ADHRS operation and determined to remain within code limits. Therefore, addition of the ADHRS will not significantly increase the probability of failure of this pipe.

Relative to a LOCA caused by inadvertent drainage of the reactor vessel (i.e., a system alignment that allows either gravity or pumped flow from the vessel via an existing isolation point), the ADHRS design and its accompanying procedural requirements make this a no more probable event than that associated with existing plant systems.

The probability of a LOCA type event is not significantly increased by the addition of the ADHRS.

Internally Generated Missiles

The ADHRS does not pose any additional hazards in this regard as a source of missiles. The ADHR is not postulated to be a source of pressurized component missiles, and potential rotating component missiles (from the pumps and air handling unit fan) are also not required to be postulated.

Reactor Coolant Pressure Boundary Overpressure

When in operation, the ADHRS pumps reactor coolant to the reactor vessel. The only time when ADHRS could be operated that reactor pressure could potentially increase is in operational condition 4 with the reactor vessel head installed. However, the maximum discharge pressure of the ADHRS is well below reactor vessel design pressure. The ADHRS otherwise will not interfere with vessel overpressure protection functions so that the probability of an event of this kind is not increased.

Fire

The ADHRS does not involve the addition of any combustible loading in excess of that assumed in the FHA to the areas in which it is located as contained in the GGNS Fire Hazards Analysis Report and thus does not contribute to the increased probability of a fire.

Loss of Electrical Power

The only Class 1E electrical loads involved with the ADHRS design are the motor operators for valves E12F066A and B. These power supplies are designed in accordance with the various Class 1E criteria described in UFSAR Section 8. The power supplies to these motors are adequate for the additional loading. Thus, there is no additional potential for loss of Class 1E power. The added non-Class 1E loads have been designed to standards consistent with the existing design for this type of load. Therefore, to the extent that loss of non-Class 1E power might initiate a plant transient, no greater probability is indicated.

Offsite Radioactive Release

The ADHRS design provides a boundary, within the ADHRS heat exchangers, between radioactively contaminated reactor coolant and the PSW system which can discharge to the environment. The boundary (heat exchanger tubes) is designed and constructed to ASME Section III, Class 3 standards and to Seismic Category I criteria. As such the probability of its failure is no greater than other plant systems containing radioactive fluids.

Inadvertent Operation of RHR Shutdown Cooling

This event is discussed in UFSAR Section 15.1.6 and relates to a slow decrease in moderator temperature leading to an increase in reactor power, when the reactor is critical or near critical. The only cause for this event cited in the UFSAR is misoperation of the cooling water controls for the RHR heat exchangers. Inasmuch as the cooling water controls for the ADHRS are no more prone to failure than those of the RHR (both are manually operated with no automatic activation or automatic control), and there would be no reason for greater operator error, the ADHRS would not significantly increase the probability of this event.

b) The ADHRS design (including operating requirements) does not significantly increase the consequences of an accident previously evaluated in the UFSAR as discussed below.

The assessment relative to consequences results in specific consideration of the following events for significant increases in consequences.

- o External high wind/tornado, tornado missile, and flooding events
- o Seismic events
- o Pipe break or leak
- o Internally generated missiles
- o LOCA
- o Hydrodynamic events
- o Loss of electrical power or instrument air
- o Fire
- o Offsite radioactive release

Other events/accidents are excluded from detailed consideration since there are no identified functional interactions or spatial/environmental interactions.

External high wind/tornado missile and flooding

Components added and employed by the ADHRS are completely housed within existing safety-related portions of the plant (Auxiliary Building) and as such are not subject to damage or other consequences of these extreme environmental effects which are discussed in UFSAR Sections 3.3, 3.4 and 3.5.

Seismic Events

The ADHRS and its components (including existing plant equipment used for the ADHRS) are designed or analyzed for safe shutdown earthquake (SSE) design basis loads in order to preserve the pressure integrity of safety-related components, the operability of the motor operators and conduit for valves E12F066A and B, and to ensure that non-safety related components of the ADHRS do not pose a hazard to safety-related plant components during a seismic event. Existing components connected to the ADHRS have been re-evaluated for the new attachment loadings. This re-evaluation also includes assurance of continued operability of any affected active safety-related components (e.g., motor operated valves) during or after the seismic event as may be required.

The structural integrity of the Auxiliary Building due to loads imposed by the ADHRS is adequate. The major ADHRS components are located on the building basemat at El. 93', and there is no appreciable effect on structural seismic response. Loads imposed on walls, beams, or other structural elements in the ADHRS design are acceptable.

Pipe Break/Leak

The ADHRS does not present hazards (pipe whip, jet impingement, etc.) from high energy line breaks (HELB) and is not subject to damage from HELBs in other plant systems.

The ADHRS safety functions are not impacted by spray or flooding from moderate energy pipe cracks in other plant systems (no such events are required to be postulated during operational conditions 4 and 5 when the ADHRS is in operation). The ADHRS also does not pose a hazard to other safety-related plant components due to spray effects of a pipe crack. Existing drainage and detection capabilities are adequate for any flooding that may be involved in the event of an ADHRS line crack or break.

Therefore, the addition of the ADHRS does not significantly increase the consequences of pipe break as discussed in UFSAR Section 3.6.

Internally Generated Missiles

The potential for damage to the ADHRS, and therefore increased accident consequences due to internally generated missiles (valve stems, rotating equipment, etc.) or turbine missiles has been assessed in accordance with UFSAR Sections 3.5.1.3 and 3.5.1.4. No additional consequences were determined to occur.

LOCA

Postulations concerning a LOCA in reactor operational conditions 4 and 5 are not contained in the UFSAR, and such an event is not analyzed (UFSAR Sections 6.2 and 15.6 deal with the design basis LOCA occurring during reactor operation).

However, to the extent that the ADHRS could either interfere with the mitigation of such an event or be the cause of the event, no increased consequences have been identified.

The controls interaction and system interaction evaluation completed for the ADHRS indicate that no interference by the ADHRS in emergency core cooling system (ECCS) functions or vessel level detection functions would occur. Protection against inadvertent drainage during operation of ADHRS is equivalent to that provided for existing plant systems through the use of interlocks and restricting operating procedures. Additionally, a failure of the ADHRS (i.e., a moderate energy pipe crack) is not required to be postulated during times when it is conveying reactor coolant (UFSAR Section 3.6A.1.1.C).

On the above basis, the ADHRS modification would not significantly increase the consequences of a LOCA.

Hydrodynamic Load Events

Such events include LOCAs (UFSAR Section 15.6), inadvertent SRV opening (UFSAR 15.1.4), and safety-relief valve operation incident to other transients or accidents. While the ADHRS would not be in operation during such events (as there is no reactor pressure source in operational conditions 4 or 5), it is nevertheless designed for such loads. Also connecting equipment was reanalyzed to the extent that these loads may be transmitted to the ADHRS, to ensure that the ADHRS would not be a hazard and thereby increase the consequences of such events.

Loss of Electrical Power or Instrument Air

The ADHRS has no safety-related function relevant to the loss of AC power as discussed in UFSAR Section 15.2.6 or instrument air as discussed in UFSAR Section 15.2.10 and no reliance on electrical power or instrument air is made by ADHRS for other plant events or accidents to ensure that consequences are not increased beyond those already evaluated.

Fire

The existing fire detection and suppression equipment is adequate for any potential fire originating within the ADHRS or incident to its presence.

Other than the motor-operators for valves E12F066A and B, and associated power and control cable, the interlock for valve E12F004C, and the pump start permissive bypass switches for valves E12F066A and B no other safety-related components are susceptible to damage by fire. To the extent that the functional capability of the E12F066A or B operators may be lost due to a fire, no increase in accident consequences is indicated. This is because the events for which they are provided (e.g., LOCAs) are not postulated to occur coincident with a fire, as indicated in the GGNS Fire Hazards Analysis Report and 10CFR50 Appendix R. The isolation function of the valves is only required during operational conditions 4 and 5. As such the ability to bring the plant to a cold shutdown condition in the event of a fire is unaffected by their operability.

Fire-induced failure of the added interlock between valve E12F004C and the RHR 'C' pump may prevent the RHR 'C' pump from starting. This is not a concern, however, as the RHR 'C' train is not required for safe shutdown of the plant incident to a fire (it provides a LPCI function only in mitigation of a LOCA). Failure of the added RHR 'A' and/or 'B' pump start permissive bypass switches for valves E12F066A and B due to a fire may prevent the RHR 'A' and/or 'B' pump from operating with suction from the spent fuel pool. This is also not a concern since the associated spent fuel pool cooling assist mode of operation of the RHR system is not required for safe shutdown of the plant incident to a fire.

The installation requirements for the ADHRS preserve the integrity of fire barriers and penetration seals so existing plant fire containment features are unaffected.

On the basis of the above, no significantly increased consequences of a fire are indicated by installation of the ADHRS.

Offsite Radioactive Release

The potential consequences of an offsite release of radioactively contaminated water due to an ADHRS heat exchanger tube failure have been shown to be within regulatory guidelines and limits and less than comparable events evaluated in the UFSAR, Section 15.7.2.

Inadvertent Operation of RHR Shutdown Cooling

As discussed in UFSAR Section 15.1.6, the effect of inadvertent operation of RHR is to slowly decrease reactor coolant temperature. Since the heat removal capacity of the RHR system is greater than that of ADHRS, the ADHRS cannot produce a greater effect. The mitigation of the event is a high neutron flux reactor scram, which would be unaffected by the installation or use of the ADHRS.

c) The function of the PSW discharge line radiation monitor is to detect substantial intersystem leakage of reactor coolant into the PSW system. This monitor and its associated alarm performs no automatic accident mitigation function or safety-related function. The PSW radiation monitor is not required to function in order to prevent accidental offsite doses from being exceeded. The design and installation of the PSW radiation monitor will be done in accordance with appropriate codes and standards to ensure that interfacing system requirements are not compromised.

d) Currently, valves E12F066A and B perform no accident mitigation function and are not required to open or close in order to bring the plant to cold shutdown conditions. The function of the two new motor operated valve thermal overload protection devices is to protect the motor operators under overload conditions. These motor operators perform no automatic or remote-manual accident mitigation function. The failure of the motor operators will have no f ctional effect on ECCS system operation. The ECCS system is capable of delivering required system flow rates with E12F066A and B failed in the open position. The failure of the motor operator to close or to be closed would prevent the establishment of an ASME Section III Class 2 pressure boundary for ECCS. However, the ECCS system has been evaluated with the failure of these valves in the open position and was found to be capable of continuing to deliver the required ECCS system flow rates. In addition, the piping beyond valves E12F066A and B is designed for process conditions that would occur in the LPCI mode. The only function of the motor operators is to allow remote-manual operation of the valves from the control room. The addition of the motor operators and the thermal overload devices will not change or affect the valves' present function or performance.

Therefore, the addition of the ADHRS and its associated PSW radiation monitor and thermal overload devices for valves E12F066 A and B would not significantly increase the probability or consequences of an accident previously evaluated.

II. These changes would not create the possibility of a new or different kind of accident from any previously evaluated.

a) ed on the functional interactions of the ADHRS, the following potential events have been evaluated with respect to the creation of a new or different type of accident from any previously evaluated:

o Loss of fuel pool cooling/fuel pool water inventory

o Offsite radioactive release

<u>Loss of Fuel Pool Cooling/Fuel Pool Water Inventory</u>

The ADHRS can be connected to the spent fuel pool either in a flush/fill mode or as a suction source for cooling when the reactor cavity/upper containment pool is filled. The ADHRS is not specifically designed for or otherwise expected to assume heat loads imposed by the spent fuel pool. Additionally, the loss or failure of the ADHRS cannot prevent the RHR system from operating in the fuel pool cooling mode. Thus, loss of the ADHRS cannot involve a loss of spent fuel pool cooling.

The operation of the ADHRS with suction on the spent fuel pool will not cause a loss of the spent fuel pool cooling system, as described in UFSAR Section 9.1. There are no mechanical, controls, or electrical interconnections between the two systems and hydraulic-type effects have been evaluated and concluded to be non-interfering.

When connected to the spent fuel pool there is potential for leakage from the pool due to a failure in ADHRS piping. This piping is designed and qualified per the applicable criteria including new operating conditions for existing piping and is no more susceptible to a break or leak than existing piping. The potential for inadvertently draining the spent fuel pool when the ADHRS is taking suction from it has been evaluated. The ADHRS uses an existing suction point on the spent fuel pool. As explained in UFSAR Section 9.1, all connections to the pool are designed to preclude drainage below a level sufficient for adequate shielding. The ADHRS use of this existing line would not interfere with this feature. Hydraulic operating effects that could lower the spent fuel pool level have been evaluated and found not to present a potential hazard relative to the shielding function of the fuel pool inventory.

The drainage potential and hydraulic effects of ADHRS operation on the spent fuel pool are consistent with those existing when operating the RHR system in the fuel pool cooling assist mode. Since the ADHRS configuration offers no less protection in this regard, it is concluded that a new or different accident type is not created.

<u>Offsite Radioactive Release</u>

The ADHRS design provides a boundary, within the ADHRS heat exchangers, between radioactively contaminated reactor coolant and the PSW system which can discharge to the environment. The boundary (heat exchanger tubes) is designed and constructed to ASME Section III and Seismic Category I requirements. As such the probability of an ADHRS boundary failure is no greater than other plant systems containing radioactive fluids (e.g. FPCCU, RWCU, or radwaste tanks).

The GGNS UFSAR Section 15.7.2 addresses a postulated unexpected and uncontrolled release of radioactivity due to a radioactive liquid waste system failure. The postulated gross failure of the ADHRS pressure boundary and the UFSAR Section 15.7.2 event are considered similar kinds of accidents in that both are unexpected and uncontrolled releases of radioactive material to the offsite boundary.

b) The PSW radiation monitor will perform no automatic accident mitigation function and will initiate no safety-related or nonsafety-related systems. The function of the radiation monitor is to detect substantial intersystem leakage of reactor coolant into the PSW system. The design and installation of the PSW radiation monitor will be done in accordance with appropriate codes and standards to ensure interfacing system requirements are not compromised.

c) The addition of the motor operators and the thermal overload protection devices on valves E12F066A and B does not change the valves original function. These valves will continue to perform no safety-related function other than serving as system code boundary classification break. The failure of E12F066A or B to close or to be closed by the operator defeats the establishment of an ASME Section III Class 2 pressure boundary for ECCS as recommended in Regulatory Guide 1.29 and committed to in UFSAR Section 3.2. However, this failure has no functional effect on the operation of ECCS. The ECCS systems have been evaluated with the failure of these valves in the open position and was found to be capable of continuing to deliver the required ECCS flow rates. In addition, the piping beyond these valves is safety-related (Class 3), Seismic Category I and designed for the process conditions that would occur in the LPCI mode.

Therefore, the addition of the ADHRS and its associated PSW radiation monitor and thermal overload devices for valves E12F066 A and B would not create the possibility of a new or different kind of accident from any previously evaluated.

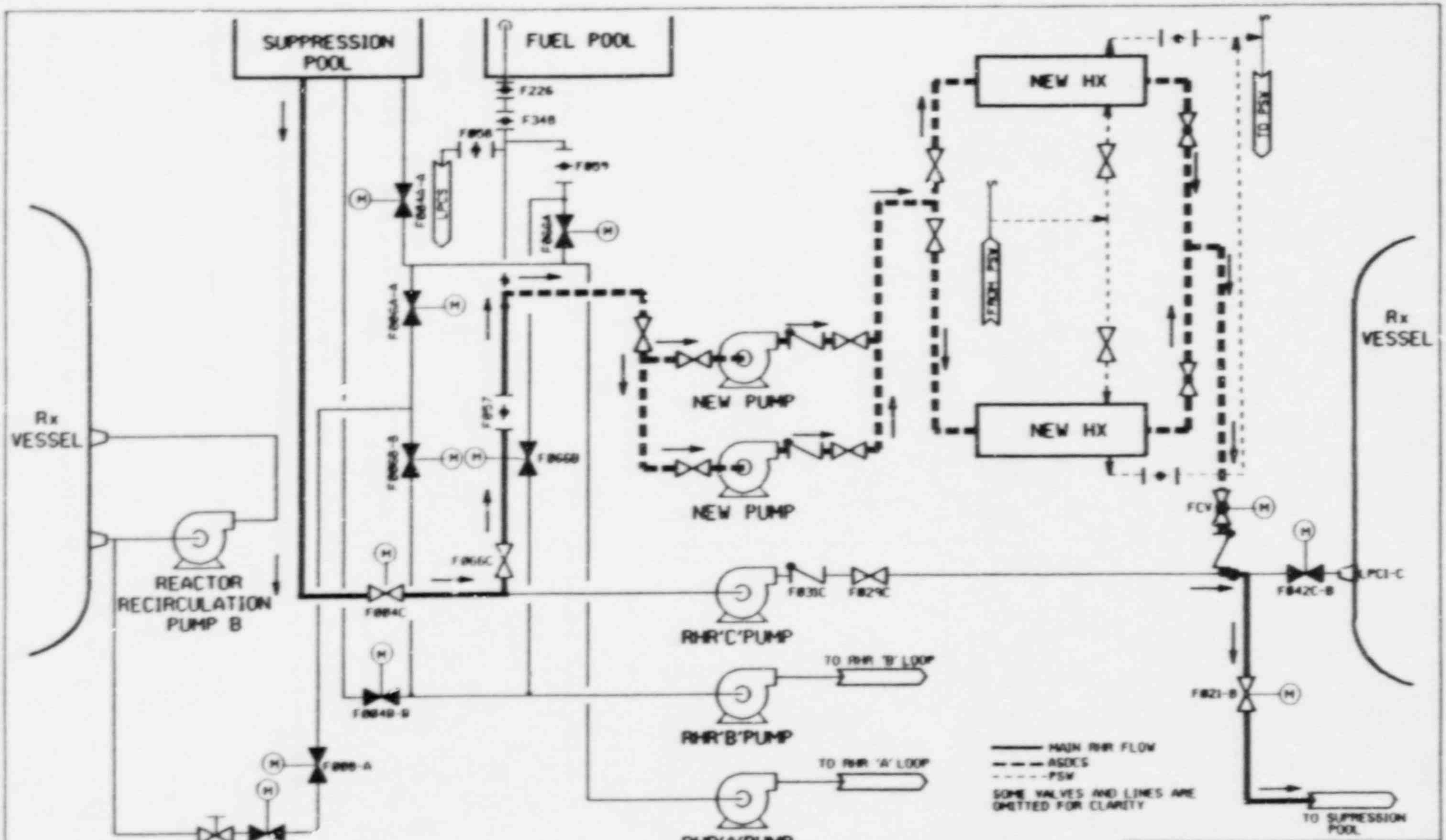
III. These changes would not involve a significant reduction in the margin of safety.

a) The ADHRS provides an improved alternate method of decay heat removal and coolant mixing as required by the bases for Technical Specification 3/4.4.9. Since the ADHRS is specifically designed to maintain the average reactor coolant temperature less than or equal to 200°F in cold shutdown and less than or equal to 140°F in refueling, the margin of safety to fulfill the requirements of Technical Specification 3/4.4.9 and Table 1.2 is maintained.

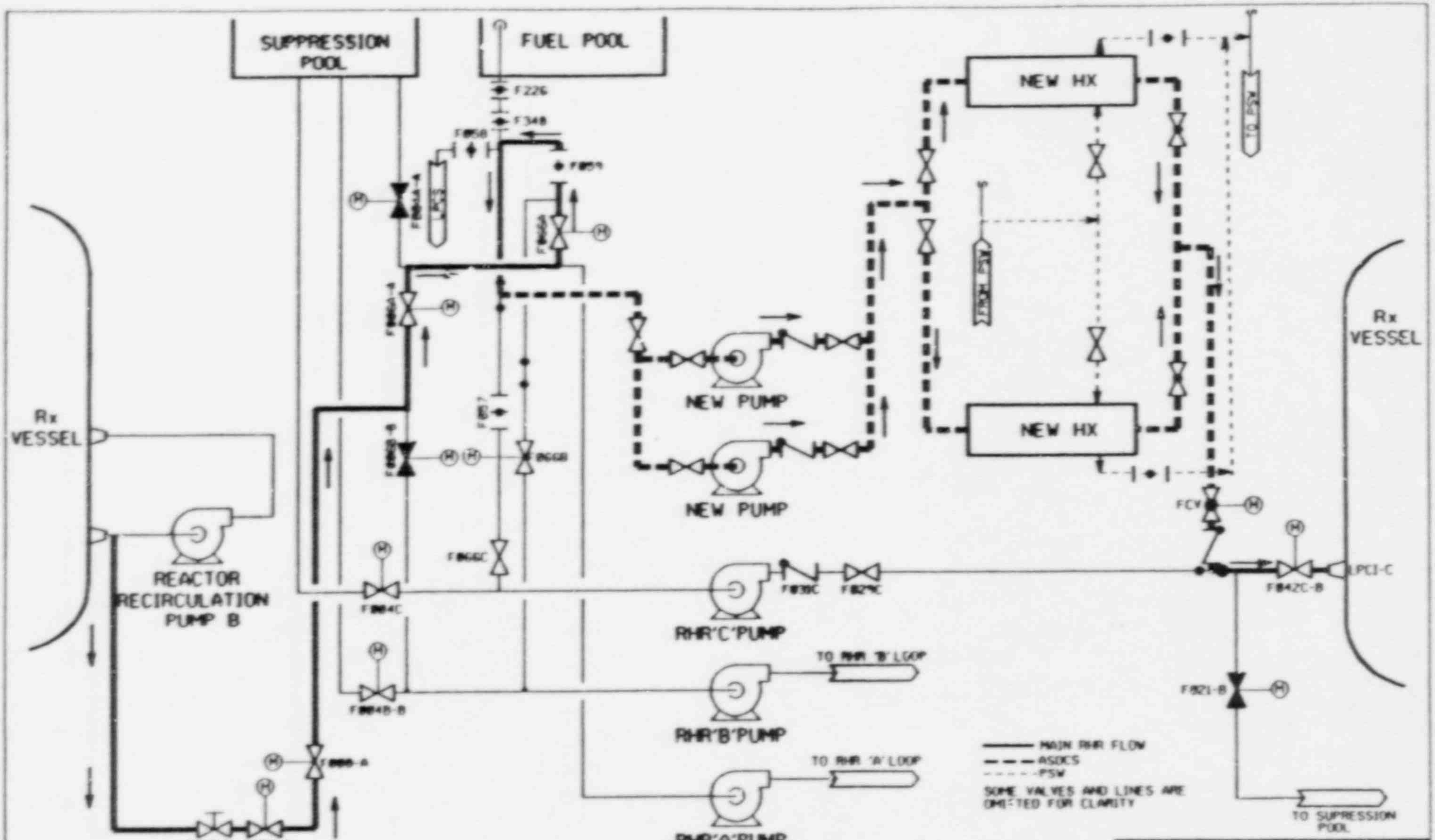
b) The ADHRS is designed and constructed to ASME Section III, Class 2 and Class 3 standards and to Seismic Category I criteria. As such, the margin of safety this system provides is equal to or greater than other comparable plant systems containing radioactive fluids (e.g. FPCCU, RWCU, etc.). The ADHRS and its associated PSW radiation monitor will have no direct or indirect impact on existing safety-related or nonsafety-related systems and thus will not affect operation of any equipment required to mitigate an accident.

c) The addition of the motor operators and associated thermal overload protection devices to valves E12F066A and B will not affect the operation or intended function of these valves. As previously noted, these valves perform no accident mitigation function. The current function of these valves is to align RHR 'A' or 'B' loops in the spent fuel pool cooling assist modes and to serve as code boundary classification breaks between ASME Section III Class 2 and Class 3 piping. The failure of these valve actuators will have no functional effect on the ECCS system.

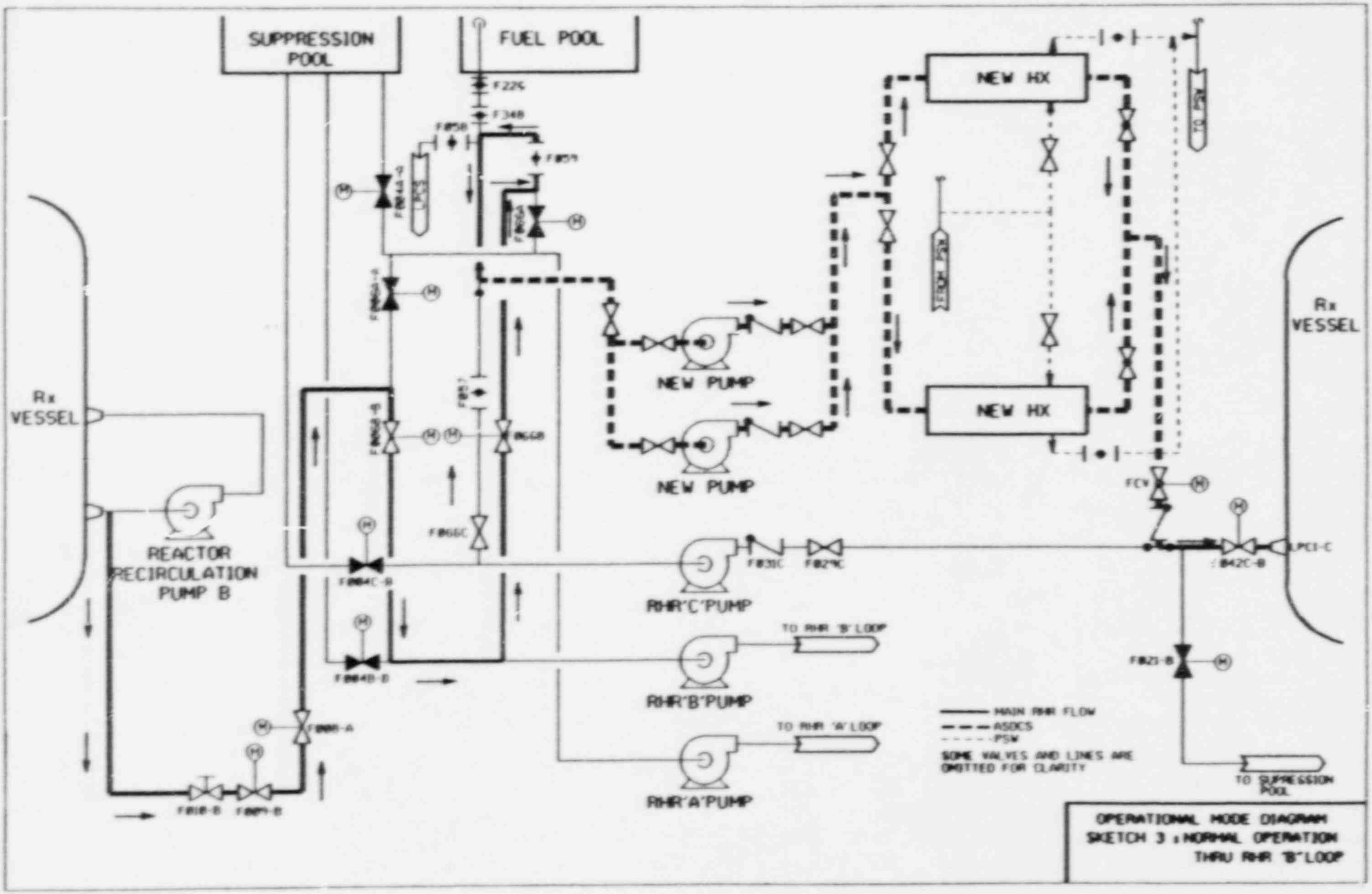
Therefore, the addition of the ADHRS and its associated PSW radiation monitor and thermal overload devices for valves E12F066 A and B would not involve a significant reduction in the margin of safety.

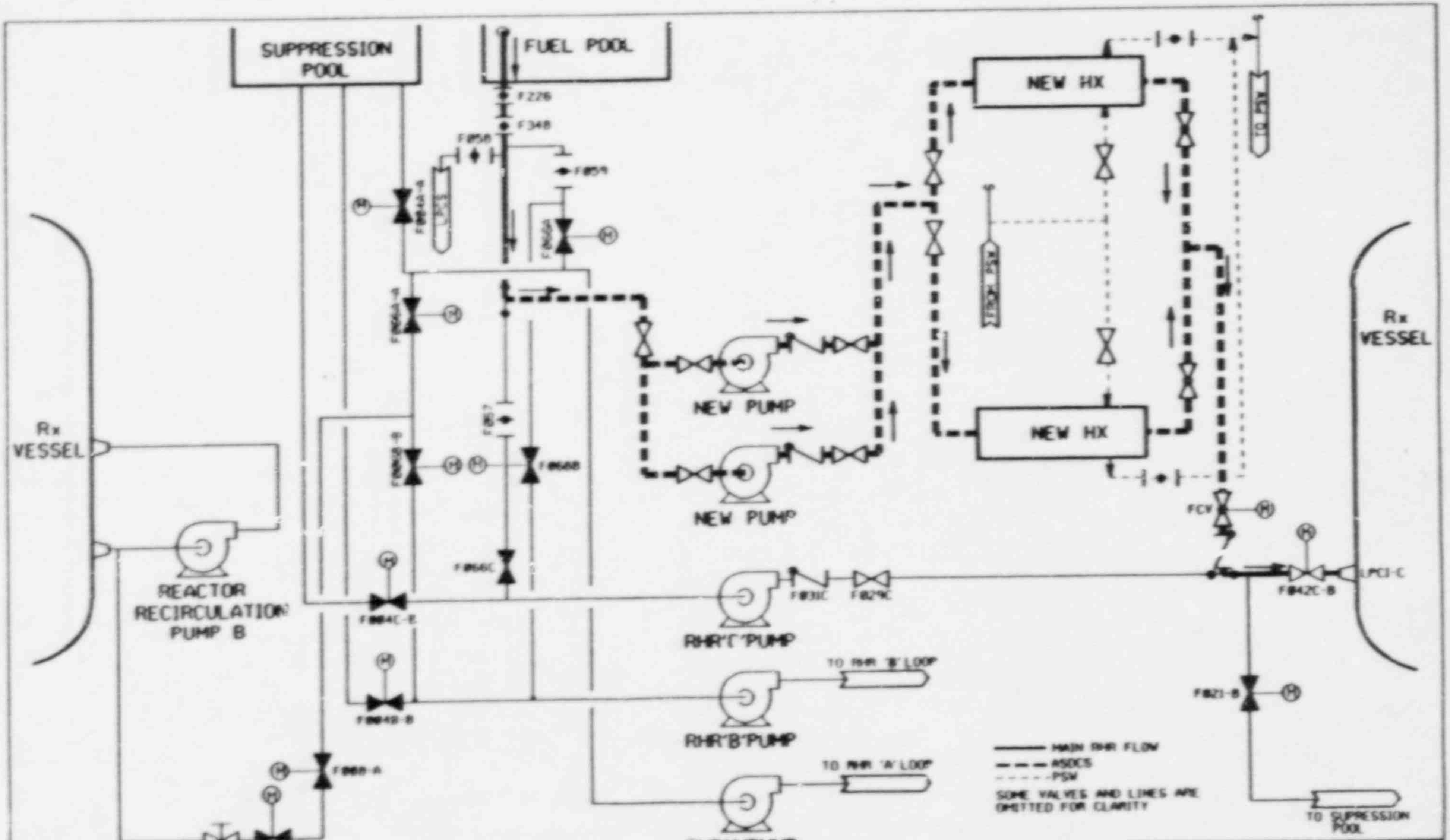


OPERATIONAL MODE DIAGRAM
SKETCH 1: TEST MODE SUCTION FROM SUPPRESSION POOL



OPERATIONAL MODE DIAGRAM
SKETCH 2 : NORMAL OPERATION
THRU RHR "A" LOOP


SUPPRESSION POOL
FUEL POOL
F226
F348
F050
F059
Rx VESSEL
REACTOR RECIRCULATION PUMP B
NEW HX
NEW HX
NEW PUMP
NEW PUMP
RHR'C'PUMP
RHR'B'PUMP
RHR'A'PUMP
F066C
F068
F004C-B
F004B-B
F008-A
F010-B
F009-B
F031C
F029C
FCV
F042C-B
LPCI-C
F021-B
TO RHR 'B' LOOP
TO RHR 'A' LOOP
TO SUPPRESSION POOL
MAIN RHR FLOW
ASDCS
PSW
SOME VALVES AND LINES ARE OMITTED FOR CLARITY
OPERATIONAL MODE DIAGRAM
SKETCH 3 : NORMAL OPERATION
THRU RHR 'B' LOOP



OPERATIONAL MODE DIAGRAM
SKETCH 4 : SUCTION FROM FUEL POOL

GRAND GULF-UNIT 1

TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. Component Cooling Water Radiation Monitor	1	At all times	$\leq 1 \times 10^5$ cpm/NA	10 to 10^6 cpm	70
2. Standby Service Water System Radiation Monitor	1/heat exchanger train	1, 2, 3, and*	$\leq 1 \times 10^5$ cpm/NA	10 to 10^6 cpm	70
3. ~~[DELETED]~~ Plant Service Water System Radiation Monitor	1	##	$\leq 1 \times 10^5$ cpm/NA	10 to 10^6 cpm	70
4. [DELETED]					
5. Carbon Bed Vault Radiation Monitor	1	1, 2	$\leq$ 2 x full power background/NA	1 to 10^6 mR/hr	72
6. Control Room Ventilation Radiation Monitor	2/trip system(h)	1,2,3,5 and**	$\leq$4 mR/hr/ $\leq$5 mR/hr#	10^{-2} to 10^2 mR/hr	73
7. Containment and Drywell Ventilation Exhaust Radiation Monitor	2/trip system(h)	At all times	$\leq$2.0 mR/hr/ $\leq$4 mR/hr(b)#	10^{-2} to 10^2 mR/hr	74
8. Fuel Handling Area Ventilation Exhaust Radiation Monitor	2/trip system(h)	1,2,3,5 and**	$\leq$ 2mR/hr/ $\leq$4 mR/hr(d)#	10^{-2} to 10^2 mR/hr	75
9. Fuel Handling Area Pool Sweep Exhaust Radiation Monitor	2/trip system(h)	(c)	$\leq$ 18 mR/hr/ $\leq$35 mR/hr(d)#	10^{-2} to 10^2 mR/hr	75

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TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
10. Area Monitors					
a. Fuel Handling Area Monitors					
1) New Fuel Storage Vault	1	(e)	$\leq$2.5 mR/hr/NA	10^{-2} to 10^{3} mR/hr	72
2) Spent Fuel Storage Pool	1	(f)	$\leq$2.5 mR/hr/NA	10^{-2} to 10^{3} mR/hr	72
3) Dryer Storage Area	1	(g)	$\leq$2.5 mR/hr/NA	10^{-2} to 10^{3} mR/hr	72
b. Control Room Radiation Monitor	1	At all times	$\leq$0.5 mR/hr/NA	10^{-2} to 10^{3} mR/hr	72

* With RHR heat exchangers in operation.

** When irradiated fuel is being handled in the primary or secondary containment.

\# Initial setpoint. Final Setpoint to be determined during startup test program. Any required change to this setpoint shall be submitted to Commission within 90 days after test completion.

With ADHR heat exchangers in operation.

(a) Trips system with 2 channels upscale-Hi Hi Hi, or one channel upscale Hi Hi Hi and one channel downscale or 2 channels downscale.

(b) Isolates containment/drywell purge penetrations.

(c) With irradiated fuel in spent fuel storage pool.

(d) Also isolates the Auxiliary Building and Fuel Handling Area Ventilation Systems.

(e) With fuel in the new fuel storage vault.

(f) With fuel in the spent fuel storage pool.

(g) With fuel in the dryer storage area.

(h) Two upscale Hi Hi, one upscale Hi Hi and one downscale, or two downscale signals from the same trip system actuate the trip system and initiate isolation of the associated isolation valves.

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Component Cooling Water Radiation Monitor	S	M	A	At all times
2. Standby Service Water System Radiation Monitor	S	M	A	1, 2, 3, and*
3. ~~[DELETED]~~ Plant Service Water System Radiation Monitor	S	M	A	#
4. [DELETED]				
5. Carbon Bed Vault Radiation Monitor	S	M	A	1, 2
6. Control Room Ventilation Radiation Monitor	S	M(a)	A	1, 2, 3, 5 and**
7. Containment and Drywell Ventilation Exhaust Radiation Monitor	S	M	A	At all times
8. Fuel Handling Area Ventilation Radiation Monitor	S	M	A	1, 2, 3, 5 and**
9. Fuel Handling Area Pool Sweep Exhaust Radiation Monitor	S	M	A	(b)
10. Area Monitors				
a. Fuel Handling Area Monitors				
1) New Fuel Storage Vault	S	M	R	(c)
2) Spent Fuel Storage Pool	S	M	R	(d)
3) Dryer Storage Area	S	M	R	(e)
b. Control Room Radiation Monitor	S	M	R	At all times

* With RHR heat exchangers in operation.

** When irradiated fuel is being handled in the primary or secondary containment.

(a) The CHANNEL FUNCTIONAL TEST shall demonstrate that control room annunciation occurs if any of the following conditions exist.
1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not in Operate mode.

(b) With irradiated fuel in the spent fuel storage pool.

(c) With fuel in the new fuel storage vault.

(d) With fuel in the spent fuel storage pool.

(e) With fuel in the dryer storage area.

\# With ADHR heat exchangers in operation.

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (CONTINUOUS) (ACCIDENT CONDITIONS) (NO)	SYSTEM(S) AFFECTED
Q1E12F074A	Continuous	RHR System
Q1E12F026A	Continuous	RHR System
Q1E12F082A	No	RHR System
Q1E12F082B	No	RHR System
Q1E12F290A	Continuous	RHR System
Q1E12F047A	Continuous	RHR System
Q1E12F027A	Continuous	RHR System
Q1E12F073A	Continuous	RHR System
Q1E12F346	Continuous	RHR System
Q1E12F024A	Continuous	RHR System
Q1E12F087A	Continuous	RHR System
Q1E12F048A	Continuous	RHR System
Q1E12F042A	Continuous	RHR System
Q1E12F004A	Continuous	RHR System
Q1E12F003A	Continuous	RHR System
Q1E12F011A	Continuous	RHR System
Q1E12F053A	Continuous	RHR System
Q1E12F037A	Continuous	RHR System
Q1E12F028A	Continuous	RHR System
Q1E12F064A	Continuous	RHR System
Q1E12F290B	Continuous	RHR System
Q1E12F004C	Continuous	RHR System
Q1E12F021	Continuous	RHR System
Q1E12F064C	Continuous	RHR System
Q1E12F042C	Continuous	RHR System
Q1E12F048B	Continuous	RHR System
Q1E12F049	Continuous	RHR System
Q1E12F037B	Continuous	RHR System
Q1E12F053B	Continuous	RHR System
Q1E12F074B	Continuous	RHR System
Q1E12F042B	Continuous	RHR System
Q1E12F064B	Continuous	RHR System
Q1E12F096	Continuous	RHR System
Q1E12F094	Continuous	RHR System
Q1E12F006B	Continuous	RHR System
Q1E12F011B	Continuous	RHR System
Q1E12F052B	Continuous	RHR System
Q1E12F047B	Continuous	RHR System
Q1E12F027B	Continuous	RHR System
Q1E12F004B	Continuous	RHR System
Q1E12F087B	Continuous	RHR System
Q1E12F003B	Continuous	RHR System
Q1E12F026B	Continuous	RHR System
Q1E12F024B	Continuous	RHR System
Q1E12F028B	Continuous	RHR System
Q1E12F009	Continuous	RHR System
Q1E12F073B	Continuous	RHR System
Q1E12F066B	Continuous	RHR System

Q1E12F066A	Continuous	RHR System

ATTACHMENT 1

ALTERNATE DECAY HEAT REMOVAL SYSTEM (ADHRS)

INTERACTION EVALUATION

GRAND GULF NUCLEAR STATION

UNIT 1

ALTERNATE DECAY HEAT REMOVAL SYSTEM (ADHRS)
INTERACTION EVALUATION

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ALTERNATE DECAY HEAT REMOVAL SYSTEM (ADHRS)
INTERACTION EVALUATION

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(Continued)

ALTERNATE DECAY HEAT REMOVAL SYSTEM (ADHRS) INTERACTION EVALUATION

I. PURPOSE

This ADHRS interaction evaluation identifies the various potential operational and physical interactions between the ADHRS and existing plant safety functions and associated safety-related structures, systems and equipment. This evaluation provides documentation and specific details to support determinations in the 10CFR50.59 safety evaluation for the ADHRS modification. It also identifies whatever procedural requirements or operational restraints may be required to establish and maintain the acceptability of this modification.

The general criteria employed in this interaction evaluation include the following:

- Avoidance of adverse effects on existing safety-related systems and related plant safety functions
- Avoidance of adverse effects on safety-related functions of the ADHRS
- Avoidance of significant equipment damage irrespective of safety implications
- Avoidance of unacceptable offsite effects.

For convenience, the evaluatio 's conducted in two broad areas as discussed in Section II and III of t evaluation:

- Functional interactions betwe the ADHRS and systems with which it may operate
- Physical interactions between the ADHRS and other plant components regardless of any functional relationship

II. FUNCTIONAL INTERACTIONS

Functional interactions between the ADHRS and existing safety systems/functions are evaluated in the following areas:

- Maintenance of operability of safety-related systems and functions
- Operating modes and combinations
- Potential for inadvertent drainage
- Controls/operational interactions

A. MAINTENANCE OF OPERABILITY OF SAFETY-RELATED SYSTEMS AND FUNCTIONS

Based on the interconnections with existing systems and design basis operating modes of the ADHRS (identified in Section II.B), the following safety-related systems and functions for which there is a functional or potential functional relationship with the ADHRS are identified:

- RHR System
  - Low pressure coolant injection mode (LPCI)
  - Shutdown cooling mode (SDC)
  - Containment spray
  - Fuel pool cooling
  - Suppression pool cooling
- Spent Fuel Pool Cooling System
  - Fuel pool cooling
  - Shielding of spent fuel
- LPCS System
  - Low pressure core spray
- Nuclear Boiler System
  - Reactor coolant temperature control
  - Reactor coolant circulation
  - Reactor core shielding/reactor vessel water inventory/pressure boundary integrity

Maintenance of the operability of the above systems and functions from a functional perspective is discussed below. Other possible impacts on operability relating to potential physical interactions are discussed in Section III of this evaluation.

1. RHR System

Operability of the RHR system for the safety-related modes/functions noted above includes the following considerations:

- Maintenance of required pressure boundary integrity, including required code classifications
- Maintenance of required power supplies

- o Maintenance of necessary support functions
- o Maintenance of ability to initiate or be initiated, including necessary instrumentation, controls, and indications
- o Continuance of proper operation and avoidance of equipment damage or undesirable consequential effects after startup

The ADHRS is designed for operation in reactor modes 4 (cold shutdown) and 5 (refueling) only. In these modes, the ADHRS is designed for operation under "normal" conditions; that is, it has not been designed or evaluated for operation subsequent to an abnormal event such as a LOCA with fission product release or an ATWS event with sodium pentaborate injection to the vessel. In reactor modes 1 through 3, the ADHRS is designed for and is required to be completely isolated from and otherwise functionally non-interactive with interconnecting systems (with the exception of an interlock between the RHR 'C' pump start controls and suction valve E12F004C, and a modification to the RHR A and B pump trip circuitry which are discussed in Section II.D). This includes isolation at the ADHRS suction and discharge interconnections with closed valves (and the discharge flow control valve motor operator electrically isolated), valves E12F066A and B closed with motor operators electrically isolated, the valves for the plant service water supply to ADHRS heat exchangers closed, and electrical isolation of the ADHRS pumps. As such there is no potential for interaction between the ADHRS and other systems/functions (with the exception of the interlock noted above) in reactor modes 1, 2 or 3. Consequently, RHR or other system functions that are pertinent only to those modes are not considered further. For RHR, these include containment spray and suppression pool cooling (which may be used in reactor modes 4 or 5, but would not be a safety [accident mitigation] function in those modes). The remaining functions are LPCI, shutdown cooling, and fuel pool cooling assistance.

The various aspects of RHR system operability maintenance are discussed below.

a. <u>Pressure Boundary Integrity/Code Classification</u>

The pressure boundary of those portions of RHR that are involved in LPCI, shutdown cooling, or fuel pool cooling assistance, can be isolated from RHR loops A and B by the remote manual closure of valves E12F066A and B. RHR Loop C is isolated from the ADHRS by the automatic closure of the ADHRS discharge check valve when the RHR 'C' pump operates. (The operator, by procedure, would also close the motor-operated flow control/ isolation valve on the ADHRS discharge. However, as the motor operator and associated controls are not Class 1E, no credit is assumed.)

Isolation of the ADHRS from these pressure boundaries is not required as a consequence of a failure (pipe or break) in the ADHRS pressure boundary, as explained in Section III.C. However, as recommended in Regulatory Guide 1.29 and committed in UFSAR Section 3.2, these safety-related portions of the RHR system involved in LOCA mitigation and shutdown cooling are to be designed and constructed to the standards of the ASME B&PV Code Section III, Class 2. As the ADHRS pressure boundary is ASME Section III, Class 3, the above isolation provisions preserve this commitment.

Insofar as the isolation provisions are not redundant, a failure of valves E12F066A or B to close, or the discharge check valve to close would defeat this isolation capability. The motor operators and discharge check valve however, are designated as safety-related so that their failure to function would be the one single failure of a safety-related component required to be assumed in safety-system response to design basis events. The acceptability of these failures are further considered in Section II.D of this evaluation.

b. Power Supplies

With the exception of the power supplies to be provided for the motor operators for valves E12F066A and B, the power supplies for the ADHRS (pumps, flow control valve motor operator, air handling unit, and radiation monitor) are from non-Class 1E sources and therefore have no potential to adversely interact with the Class 1E electrical power supplies to the existing RHR system. The motor operators for valves E12F066A and B and associated controls, cables, and conduit are all designated safety-related, seismic Category 1 and Class 1E. Therefore, their potential for adverse interaction with other necessary RHR functions is no greater than for existing equipment, and is acceptable.

c. Support Functions

Functional support functions for the RHR system include pump bearing and seal cooling and room cooling. These are provided by the Standby Service Water (SSW) system. The ADHRS design does not use the SSW system for any purpose and there is no potential for adverse functional interactions.

d. Ability to Initiate Safety Functions

This is addressed in Section II.D

e. Continuance of Proper Operation/Avoidance of Damage

This is addressed in Section II.D

f. Pump Minimum Flow Capability

This is addressed in Section II.D

g. Keep-Full Capability

This is addressed in Section II.D

2. Spent Fuel Pool Functions

One design mode of the ADHRS is to take suction from the spent fuel pool and return flow to the reactor vessel via the RHR 'C' loop LPCI injection line. This mode would only be used when the reactor cavity and upper containment pool are filled and a flowpath is established with the spent fuel pool. The ADHRS may provide cooling in this instance when the common RHR SDC/ADHRS suction line from the reactor vessel is out of service for maintenance/ testing on valves E12F008 and E12F009, for example.

Operability of safety-related spent fuel functions includes the following considerations discussed below:

- Maintenance of spent fuel pool temperature limits
- Maintenance of spent fuel pool water levels

a. Spent Fuel Pool Temperatures

The Technical Specifications (Section 3/4.7.9) require that the spent fuel pool be maintained below 140 degrees F. This is normally accomplished by the spent fuel pool cooling and cleanup (FPCCU) system. As noted in UFSAR Section 9.1, the RHR system may be used to assist in this function.

There is no sharing of or interconnection of systems, components or functions between the spent fuel pool cooling system with the exception of the pool itself. This includes piping, valves, electrical or other power supplies, and support systems such as room cooling. No adverse interaction is evident when the FPCCU system and ADHRS take suction on the spent fuel pool system simultaneously. This includes any hydraulic effects because, as indicated on the system flow diagram (drawing SFD-1085, Rev. 3), the RHR system in the fuel pool cooling backup mode, which uses the same suction path as the ADHRS, operates at a maximum of 7450 gpm, compared to the maximum ADHRS flow of approximately 4000 gpm.

b. Shielding of Spent Fuel

The water level above spent fuel stored in the spent fuel pool is intended to provide shielding. The ADHRS could affect this function by either causing drainage of the pool due to a leak or lowering the pool level due to hydraulic effects.

As indicated in UFSAR Section 9.1, piping connected to the spent fuel pool is designed to prevent potential leakage from causing pool level to decrease below minimum required shielding levels. This is accomplished by location of the pipe penetration to the pool, use of siphon breakers, or other means. As the ADHRS uses existing piping for its fuel pool suction mode, this protective feature is maintained.

When operating in this mode, a hydraulic gradient is necessary for flow from the reactor cavity area through the refueling canal to the spent fuel pool, which introduces the possibility of lowering the pool level to below the minimum required for shielding. However, as the RHR system can also operate with the same flow path and at higher flows, with a correspondingly greater gradient, this circumstance is bounded by the current design.

3. LPCS System

The only functional interface between the ADHRS and the LPCS system is at the suction piping from the spent fuel pool, where a branch line is provided to the LPCS pump suction for testing. This is separated from the ADHRS flow path (in either vessel to vessel or fuel pool to vessel modes) by a normally closed, local manually operated valve. As such there is no potential for interactions between the ADHRS and the LPCS function.

4. Nuclear Boiler System

The various nuclear boiler safety-related functions relative to ADHRS use are discussed below.

a. Reactor Coolant Temperature

The ADHRS is provided to maintain reactor coolant temperature limits as required by the Technical Specifications for reactor modes 4 and 5. However, this is not a safety-related function relative to ADHRS. Use of the (functionally) non-safety related ADHRS as the operating system or required backup is provided for in Technical Specifications sections 3/4.4.9.2 and 3/4.9.11.

The cooling/temperature maintenance capacity of the ADHRS is designed for and calculated to be adequate at any time subsequent to 24 hours after reactor shutdown, based on nominal decay heat rates and PSW temperatures and flows. However, as required by the Technical Specification sections cited above, capability of the ADHRS shutdown cooling must be demonstrated prior to use.

b. Reactor Coolant Circulation

As required by the Technical Specification sections noted in Section II.A.4.a, adequate reactor coolant circulation must be maintained in the event an RHR SDC train or reactor recirculation pump is not used for forced circulation. As explained in the Technical Specification bases, this circulation is necessary to maintain accurate indication of reactor coolant temperature in the region of the core. This function can be provided by the Reactor Recirculation system pumps, subject to flow restrictions as indicated in Section II.B. The ADHRS will provide sufficient circulation by itself for discharge through the RHR 'C' LPCI injection nozzle and at the ADHRS design minimum flow of 1000 gpm.

c. Reactor Core Shielding/Vessel Water Inventory/Pressure Boundary Integrity

The ADHRS is required to support these functions in that it forms an extension of the reactor coolant pressure boundary when in operation.

This is accomplished by designing and constructing the ADHRS reactor coolant boundary components to the requirements of ASME Section III, Class 3, including all pertinent loads and load combinations, including seismic Category I loads.

In the above regard, then, it is considered that the ADHRS adequately supports nuclear boiler safety functions of vessel water inventory/pressure boundary maintenance.

B. OPERATING MODES AND COMBINATIONS

With the sharing by ADHRS of piping common to the RHR A, B and C loops, LPCS and reactor recirculation system, the possibility of simultaneous operation of any of these systems and ADHRS is examined to identify any unacceptable consequences. It should be noted that the design intent of the ADHRS does not require that any other RHR or LPCS mode other than standby be in operation when the ADHRS is in operation. The detailed design, therefore, has not assessed such conditions via calculations, etc. This evaluation only identifies those modes of other systems which may be used, from a function standpoint, with the ADHRS implemented.

The ADHRS may be in one of the following modes:

- Standby ("standby" is defined as filled and vented and aligned for operation with the exception that valves E12F066A and B, and the ADHRS discharge throttling/isolation valve are closed. Pumps and motor-operated valves are capable of remote manual actuation. PSW, the AHU and the radiation monitor may or may not be in operation.)

- Flush mode (suppression pool suction from RHR 'C' and return to the pool via RHR 'C' full flow test line)

- o RPV to RPV cooling mode (via either valves E12F006A and E12F066A or E12F006B and E12F066B)

- o Fuel pool to RPV cooling mode

Other potential modes of ADHRS such as LPCI or fuel pool cooling are not design modes of the ADHRS.

RHR trains A or B could conceivably be in one of the following modes in reactor modes 4 or 5.

- o Standby (System alignment requirements for "standby" when the ADHRS is in operation may differ from present procedures. These requirements are defined in Section II.D).

- o Shutdown cooling (SDC)

- o Suppression pool cooling/test (SPC)

- o Fuel pool cooling assist/test (FPA)

- o Inoperable

Other modes, as identified on GE process diagram 762E425BA, Rev. 8, and Bechtel System Flow Diagram are not included since they involve accident responses (LPCI, containment spray, fuel pool makeup) in which ADHRS would be secured (the ability to achieve the LPCI mode is examined in Section D), modes for which the conditions are not appropriate for times when ADHRS may be operating (2-loop shutdown cooling and steam condensing), and the intermittent loop flush mode.

RHR train C could be in either standby (as noted above for RHR 'A' and 'B', this may differ from current procedures), test from the spent fuel pool, or test from the suppression pool.

The only LPCS mode pertinent to this review is the test mode with suction from the spent fuel pool.

The reactor recirculation system may be in operation to provide required reactor coolant circulation for temperature measurement.

The various mode combinations are examined below as a function of the ADHRS mode. The acceptability of the combination is reviewed in terms of:

- o Suitability for equipment operation (e.g., adequate NPSH)

- o Acceptability of mode function

- o Potential undesirable event or accident conditions assuming a single failure/operator error

As noted in Section II.A, when the ADHRS is in a secured status there are no functional interactions with interconnecting systems; therefore, this mode is not evaluated further.

Case 1 ADHRS mode: Flush (suppression pool to suppression pool)

Reactor cavity/upper containment pool: drained or filled

Other System Modes	Acceptability
RHR A:	
Standby/Inoperable	OK
SDC	OK
SPC	OK
FPA	Not acceptable - potential to drain spent fuel pool to suppression pool via RHR 'C' suction line and/or possibly to pump suppression pool to spent fuel or upper containment pool.
RHR B:	
Standby/Inoperable	OK
SDC	OK
SPC	OK
FPA	Not acceptable - potential to drain spent fuel pool to suppression pool via RHR 'C' suction line and/or possibly to pump suppression pool to spent fuel or upper containment pool.

Other System Modes	Acceptability
RHR C:	
Standby/Inoperable	OK
SP Test	Not acceptable - adequacy of NPSH for RHR 'C' pump or ADHRS pumps has not been evaluated.
FP Test	Not acceptable - same reasons as RHR 'A' in FPA above
LPCS	
FP Test	Not acceptable - same reasons as RHR 'A' in FPA above.
Reactor Recirculation	OK

Case 2 ADHRS Mode: RPV to RPV cooling via E12F006A and E12F066A

Reactor cavity/upper containment pool: drained or filled

Other System Modes	Acceptability
RHR A:	
Standby/Inoperable	OK
SDC	Not acceptable - not analyzed for NPSH adequacy or flow split between ADHRS and RHR SDC.
SPC	Mode combination not possible, valves E12F004A (suppression pool suction) and E12F006A are interlocked such that E12F004A cannot be open when E12F006A is open
FPA	Not acceptable - not analyzed for NPSH adequacy or flow split.

Other System Modes	Acceptability
RHR B:	
Standby/Inoperable	OK
SDC	Not acceptable - NPSH adequacy and flow split not analyzed
SPC	Not acceptable - potential for inadvertent drainage of reactor vessel to suppression pool if E12F066B is opened by operator error.
FPA	Not acceptable - see RHR 'A' above.
RHR C:	
Standby/Inoperable	OK
SP Test	Not acceptable - inadvertent drainage of reactor vessel will occur through full-flow test valve E12F021C.
FP Test (cavity filled)	Not acceptable - not analyzed for adequacy or flow split.
LPCS	
FP Test (cavity filled)	Not acceptable - not analyzed for adequacy or flow split.
Reactor Recirculation	OK - Subject to maximum flow rate restrictions to ensure adequate NPSHA for ADHRS pumps.

Case 3 ADHRS Mode: RPV to RPV cooling via F006B and F066B

Reactor cavity/upper containment pool: drained or filled

This case is evaluated the same as Case 2 except the evaluations for RHR A and B trains are reversed.

Case 4 ADHRS Mode: Spent fuel pool to RPV cooling reactor cavity/upper containment pool: filled

Other System Modes	Acceptability
RHR A:	
Standby/Inoperable	OK
SDC	OK
SPC	OK
FPA	Not acceptable - not analyzed for NPSH adequacy or flow split. Potential for under- cooling of RPV or spent fuel pool.
RHR B:	
Same as RHR 'A' above.	
RHR C:	
Standby/Inoperable	OK
SP Test	Not acceptable - inadvertent drainage of spent fuel pool/upper cavity will occur through full flow test valve E12F021C.
FP Test	Not acceptable - not analyzed for NPSH adequacy or flow split.

Other System Modes	Acceptability
LPCS	
FP Test	Not acceptable - not analyzed for NPSH adequacy or flow split.
Reactor Recirculation	OK

From the above evaluations it is seen that the ADHRS and the interconnected RHR and LPCS may not be operated in certain mode combinations based on the various criteria used for acceptability. If, in a given situation, the Technical Specifications require any of the interconnected systems to be operating in one of the identified modes, ADHRS operation may be precluded if the above evaluation shows that simultaneous operation is not acceptable.

The results from the above evaluation are applicable for operation of multiple trains of interconnected systems (e.g., both loops of RHR SDC) concurrent with ADHRS. They are either acceptable or prohibited based on the evaluations given for single train operation.

C. INADVERTENT DRAINAGE

Inadvertent drainage is defined here as an event that reduces the water level above the reactor core by error or failure, but which does not involve a pressure boundary failure (i.e., pipe crack). Typically this involves establishment of a flow path through existing piping from the reactor vessel or connected bodies to a non-coolant boundary location.

To ensure that the ADHRS installation or operation will pose no greater possibility of such an event than existing systems or system alignments, an evaluation has been conducted of inadvertent drainage considering the ADHRS. It should be noted that to the extent that plant protective features may remain functional (e.g., automatic isolation of valves E12F008 and E12F009 on level three, which is not required to be operable in modes 4 or 5, but may be functional) during reactor modes 4 and 5, the possibility of actual core uncovery remains protected against regardless of the use of ADHRS as confirmed in Section II.D. Nevertheless, a design objective of the ADHRS and any accompanying operational consideration specified in the DCP is to ensure that the ADHRS does not increase the probability of such an event.

The evaluation has been conducted considering the following criteria:

- o A single active component failure or operator error must be considered. Single active failures involve a failure of a component to operate on demand and do not consider change of position without an actuating signal (i.e., "hot shorts"). Operator errors are not considered that involve two different procedural step violations.

- o Potential drainage paths of 1" diameter or less are not considered.

The general circumstances that can lead to inadvertent drainage relative to the ADHRS are the following:

- o As a consequence of simultaneous operation of the ADHRS and interconnected systems

- o As the result of improper system alignments prior to the use of ADHRS

- o As the result of the automatic or manual initiation of safety-related functions of systems interconnected with the ADHRS

The first category above is covered in Section II.B. The second is evaluated in Table 1, attached. The last is also partially covered in Table 1, and is further evaluated in Section II.D.

In Table 1, all potential drainage paths greater than 1-inch in diameter of the reactor vessel are evaluated considering the ADHRS in operation. Required procedural steps or hardware features are identified to satisfy the criteria noted above.

With the implementation of the procedural steps/ hardware features noted in Table 1, the system operation restrictions identified in Section II.B, and the restrictions noted in Section II.D, the ADHRS will present no greater potential for inadvertent reactor vessel drainage than existing systems or system alignments.

D. CONTROLS/OPERATIONAL INTERACTIONS

1. General Approach

The development of this part of the interaction evaluation was based on the following general steps:

a. Establish controls interaction acceptance criteria and evaluation assumptions

b. Identify basic plant conditions relative to ADHRS

c. Identify design basis events in above plant conditions involving potential interaction with ADHRS

d. Identify systems involved in above design basis events involving potential ADHRS interface and their safety functions

e. Identify all potential functional interface points (mechanical process, electrical, controls) between the ADHRS and the systems in 'd.' above and connecting equipment that may involve a control function

f. Identify all interlocks, remote manual controls, alarms, and indication for each item in 'e.' above (adjust events and systems in 'c.' and 'd.' above if appropriate)

g. Establish postulated initiating events/accidents from 'a.' and 'c.' above. Establish ADHRS and other system initial conditions

h. Test possible events/configurations for the various interaction acceptance criteria in accordance with evaluation assumptions

i. Evaluate results. Identify cases that are unacceptable or potentially unacceptable. Identify procedural or hardware requirements that may be necessary to maintain acceptability

2. Implementation

The specific details implementing the general steps (Section II.D.1) are discussed below.

a. Evaluation Criteria

Relative to the controls and interlocks of existing systems and the ADHRS, no interaction during design modes of operation is acceptable which could:

- o Prevent initiation of safety functions
- o Render required safety systems inoperable when in a standby mode
- o Cause failure or degradation of a safety-related system once its safety function has been initiated
- o Cause a safety-related failure or degradation of the ADHRS when a safety function is initiated
- o Potentially degrade plant safety by the operation of ADHRS alone or in conjunction with safety-related systems
- o Cause significant equipment damage or otherwise adversely impact outage schedule irrespective of safety implications

In this evaluation, the following assumptions are made:

- o In addition to an initiating event, a single active failure in the response systems (or required supporting systems) occurs or an operator error is made. Single active failures involving actuation without demand (i.e., "hot shorts") are not assumed. Single active component failures are only reviewed in general relevant to required responses of the ADHRS or interconnected systems relevant to ADHRS interties. Other failures, unrelated to ADHRS or its intercon- nections/influence on existing systems and functions are considered to have been adequately evaluated in support of the existing design.

  An operator error is to be considered a single procedural error, not a series of errors. Operator errors are only assumed relevant to a demand situation in response to an event, i.e., random errors unrelated to the event, are not considered. Errors made prior to the event (e.g., improper lineup) are to be considered.

- o Non-safety grade equipment may or may not function as designed and must be considered in both cases.
- o ECCS requirements are per the Technical Specifications. For reactor modes 4 and 5, two trains are required, but may be from the same division. For mode 5 with the cavity flooded, no

trains are required per the Technical Specifications, but one train is required to be functional per Grand Gulf Technical Specification Position Statement No. 109. In these modes, ECCS "operability" does not require automatic initiation/realignment functions to be available as long as the safety function can be remote manually initiated/ controlled from the main control room.

- o RHR Shutdown Cooling requirements are per the Technical Specifications. In mode 4, and mode 5 with the cavity unflooded, two operable trains or alternates are required. In mode 5 with the cavity flooded, one operable train or alternate is required. Activation of the RHR SDC backup train or ADHRS mode is allowed by local manual operation of system components; remote manual activation is not required.

- o No pre-assumptions are made regarding ADHRS alignments relative to operable ECCS trains (i.e., even though the B RHR loop may be inoperable for maintenance, ADHRS flow may be routed through a portion of B loop piping in this evaluation).

- o Simultaneous events (such as a LOCA and a high pressure event) are not considered unless there is a causal relationship between the events. Concurrent events (such as LOCA and loss of offsite power [LOP]) are also not considered.

b. <u>Basic Plant Conditions</u>

The purpose of the ADHRS results in two basic plant conditions for evaluation of ADHRS interactions:

1. Plant Operating Condition - reactor modes 1, 2 and 3. ADHRS is secured and isolated at all connections with safety-related systems. Pump motors and valves E12F066A and B motor operator breakers electrically isolated.

2. Plant Shutdown Condition - reactor modes 4 and 5. ADHRS may be in operation or aligned for operation. Support systems may be operating (Room cooler and radiation monitor).

c. <u>Design Basis Events to be Considered</u>

Relative to potential controls interactions, the following events are considered applicable:

- o Plant operating condition - LOCA, ATWS, etc.

- o Plant shutdown condition

  - LOCA (inadvertent drainage)
  - Loss of decay heat removal
  - High reactor pressure (mode 4 with head on)

Other events or accidents evaluated in FSAR have been reviewed and determined to have no functional relationships with the installation or use of the ADHRS relative to any controls/operational affects. Further details are provided in the 10CFR50.59 evaluation. It should be noted that the events postulated in this evaluation for the plant shutdown condition are not necessarily design basis events as contained in the UFSAR, but are nevertheless evaluated to ensure that such a situation does not represent a potentially new hazard to plant safety.

d. Safety Systems and Functions for Design Basis Events

For the above events, the following systems and their functions that may interact with ADHRS are identified:

- o LOCA - RHR loops A, B and C in LPCI injection, containment spray, and suppression pool cooling mode for plant operating condition. LPCI injection mode only for plant shutdown condition. LPCS in injection mode.

- o Loss of decay heat removal - RHR in the shutdown cooling mode.

- o High reactor pressure (in mode 4) - High pressure/low pressure interlocks on RHR system to isolate low pressure piping.

- o Radioactive release - see discussion in Section III.E of this evaluation.

e. Potential Functional Interface Points

The following potential functional interface points have been reviewed in the controls interaction evaluation as they relate to safety system functions identified in Section II.D.2.D:

- o Process valves

- o Pumps

  - RHR A, B and C
  - LPCS
  - ADHRS
  - RHR Jockey Pumps A, B & C

- o Electrical (None)
- o Indication/alarm
  - Reactor Level
  - Reactor Pressure

The above items are listed in Table 2 along with their associated interlocks, controls, alarms, and indication.

f. <u>Postulated Initiating Events and Initial Conditions</u>

1. Plant Operating Condition (Modes 1, 2 or 3)
   - o Initiating event - LOCA, false LOCA
   - o Initial conditions - RHR A, B & C in standby ADHRS secured and mechanically and electrically isolated.
2. Plant Shutdown Condition (Modes 4 or 5)
   - o Initiating Events
     - LOCA or false LOCA (instrument failure)
     - Loss of RHR shutdown cooling (failure unspecified)
     - High reactor pressure or false high pressure (instrument failure)
   - o Initial Conditions
     - ADHRS -- Standby/Secured
       -- Flush
       -- Test-Suppression pool to suppression pool
       -- Operating-RPV to RPV via E12F006A & E12F066A
       -RPV to RPV via E12F006B & E12F066B
       -Fuel pool to RPV
     - RHR A and/or B
       -- Standby
       -- Shutdown cooling
       -- Suppression pool cooling/test
       -- Fuel pool assist
       -- Secured (inoperable)

- RHR C
  - -- Standby
  - -- Full flow test
  - -- Secured (inoperable)

- Reactor Mode
  - -- 4 (head on)
  - -- 5 - cavity drained
    - - cavity flooded

o Event Combinations

- Single failures
- Operator errors

o Operable Systems per Technical Specifications

- ECCS
- Shutdown cooling

g. <u>Evaluation of Possible Events/Configurations for Interaction Acceptance Criteria</u>

1. Identification of Possible Combinations

Certain operating combinations of the ADHRS and interconnected systems are not allowed as identified in Section II.B. These include the following:

o Operation of RHR train A, B or C in any mode other than standby or inoperable when the ADHRS is operating in the RPV to RPV cooling mode.

o Operation of RHR A or B in the spent fuel pool cooling assist mode when the ADHRS is operating in the spent fuel pool to RPV mode.

o Operation of RHR 'C' in any mode other than standby when the ADHRS mode is operating in the RPV to RPV cooling or Fuel Pool to RPV cooling mode.

o Operation of LPCS in the fuel pool pump test mode when ADHRS is operating in the RPV to RPV or fuel pool to RPV cooling mode.

The above situations are considered to be proscribed and are not evaluated further. The following possible combinations then remain:

- o ADHRS in standby/secured and RHR A or B in SDC, suppression pool cooling, or fuel pool cooling assist mode; RHR 'C' in suppression pool test or fuel pool test mode. LPCS in fuel pool test mode.

- o ADHRS in RPV to RPV cooling mode and RHR A, B and C, and LPCS in standby.

- o ADHRS in the spent fuel pool to RPV cooling mode and RHR A or B in the standby, SDC or suppression pool cooling mode.

  - ADHRS in the flush mode and RHR A or B in standby, SDC or suppression pool cooling mode.

2. Relation of Acceptance Criteria to Specific System Conditions/Capabilities

- o Initiation of safety functions

  - Pumps can start (automatically or remote manually)

  - Flow path can be established (automatically or remote manually)

  - Required flow delivered (no bypasses, short circuits)

  - Support systems operate (power, cooling, etc.)

- o Operability during standby (in addition to capability above)

  - Keep - fill capability functional

- o Failure/degradation after initiation

  - Minimum flow needs satisfied

  - Inadvertent drainage avoided

  - Code class boundary preserved

- o Degrade plant safety

  - ADHRS pressure integrity preserved

- o Significant equipment damage

  - Case-by-case evaluation of consequences

3. Evaluations

The evaluations are grouped below into those involving the plant in an operating condition (modes 1, 2 or 3) wherein the ADHRS is secured, and those involving the plant in a shutdown condition (modes 4 and 5) when the ADHRS may be operating. For the plant operating condition, the evaluation reviews only those aspects of the ADHRS design that could have an interaction when the ADHRS is secured and isolated. For the shutdown condition, certain generic aspects to any set of event conditions are first reviewed, then specific event scenarios are evaluated.

The event cases are categoried by the assumed ADHRS mode of operation and the assumed event or accident that is postulated to occur. Various subcases are identified involved assumed operable ECCS trains, reactor mode, etc. These are summarized in Table 3. The specific event evaluations are contained in Tables 4 through 9. In the tables, the following information is provided:

- Assumed initial conditions including valve lineups and operating equipment

- Sequence of events to control the event, including automatic equipment response and operator actions

- Analysis of events, including satisfaction of the various criteria above, and analysis of single active failures or operator errors.

- Conditions as to acceptability and identification of any procedural requirements necessary for a satisfactory response to the event.

o <u>Plant Operating Condition (Modes 1, 2 and 3)</u>

As indicated in Section II.D.2.f the postulated initiating event may be any of a variety of events that would involve the RHR and/or LPCS systems in their mitigation. As indicated, the ADHRS is functionally isolated (mechanically, electrically and controls/ instrumentation) with the exceptions of the added interlock to from valve E12F004C to the RHR 'C' pump and the modification of the RHR A and B pump start permissive circuitry.

As such, only this interlock and modification need be considered for any interaction effects in modes 1, 2 or 3.

The E12F004C interlock protects the RHR 'C' pump by preventing it from starting if its suction valve from the suppression pool, E12F004C, is not open. The interlock is provided as a consequence of required inadvertent drainage protection as noted in Table 1. From Table 2, it is seen that there are no other interlocks associated with valve E12F004C, so there is no possibility of an adverse interaction of interlocking functions. The interlock in and of itself does not impact the RHR 'C' keep-fill capability, minimum flow bypass function, system flow path (other than valve E12F004C), or any support systems of the RHR 'C' loop.

In the normal, low pressure injection mode lineup of RHR 'C' in modes 1, 2 and 3 valve E12F004C is open to support automatic system initiation in the event of a LOCA. The only potential for an adverse effect of this Class 1E interlock is a failure in which the pump is prevented from starting even when valve E12F004C is open. While this event would defeat the RHR 'C' loop LPCI function for a LOCA, no impact on the adequacy of the overall plant response is indicated since no other single failures of safety-related components would need to be assumed, and the above situation would be within the bounds of current analyses, as indicated in UFSAR Section 6.2. To the extent that such a failure may contribute to the probability of failure of the RHR 'C' train, this is considered to be counterbalanced by the decreased potential for an automatic pump start and possible failure due to valve E12F004C being inadvertently left closed.

The ADHRS design includes a modification to the pump operating permissive circuitry of the RHR A and B pumps. As seen in Table 2, the pumps trip if a suction path is not available, i.e., the SDC suction path (E12F008, E12F009 and E12F006A/B), the LPCI suction path (E12F004A/B), and the fuel pool suction path (E12F066A/B)are closed. If all of these paths are closed, the pumps trip immediately upon being started.

In the normal ADHRS vessel-to- vessel cooling modes E12F066A or B would be open, and the balance of the fuel pool suction path would be closed. As such, the RHR A or B pump would not be inhibited from operating as long as E12F066A/B was open and regardless of whether an actual suction path was present. This presents a problem if a LOCA/inadvertent drainage event (or false indication of one) occurs with the ADHRS in operation. In this case, if the SDC suction source is automatically isolated on reactor level 3, and the pumps are started on a level 1 signal, they would operate without a suction source and could suffer damage. Other scenarios of operator action with or without automatic actuations and only one operator error could lead to the same result. Depending on the initiating circumstances (actual or false low level indications) this could result in a loss of ECCS capability (with a single failure in the redundant train) or significant economic consequences due to damage of an RHR pump.

To preclude such an event, a Class 1E switched bypass of the E12F066A/B permissive portion of the circuit is to be added. With the bypass on, the pumps will trip if only E12F066A/B is open and neither of the other two suction paths are open. This bypass does not affect any safety system response identified in the UFSAR including any plant design basis accident indicated in Chapter 15 of the UFSAR. Use of valves E12F066A or B (other than as planned for ADHRS) is only for RHR A or B suction from the spent fuel pool when assisting the spent fuel pool cooling system as indicated in UFSAR Section 9.1, or in a pump test mode as shown on the system flow diagram. In neither case is automatic actuation involved, or remote-manual actuation required. Currently E12F066A and B are unpowered, locally manually operated valves. If such modes were required at any time to be used, the bypass can be switched off and, after the proper alignment of all valves in the suction path, the pumps can be operated with the LPCI and SDC suction paths closed.

The above scheme does not increase the probability or consequences of any accident previously evaluated in the UFSAR (as no accident involving the failure of or use of E12F066 is indicated in the UFSAR), nor does it increase the probability of a malfunction of equipment important to safety or the consequences of the malfunction (as the bypass does not interact with necessary permissives for the RHR pump to operate when E12F004A/B is open for LPCI, or when the vessel suction path valves are open for SDC).

- <u>Plant Shutdown Condition (Modes 4 and 5)</u>

For the variety of potential operating modes and event scenarios, the interaction evaluation needs to consider the following aspects:

- Initial plant condition

  -- Reactor mode
  -- Cavity drained/flooded
  - ECCS required
  -- ECCS operable
  -- SDC required
  -- SDC operable/in operation
  -- ADHRS mode
  -- RHR mode

- Initiating event

- Assumed single failure/operator error

- Assumed ADHRS response (i.e., failure of pump to turn off, low suction interlock to function, discharge valves to close)

- Sequence of events/operator actions

- Evaluation against acceptance criteria noted in Section II.D.2.g.2

Certain aspects of the above can be considered in a generic sense and assessed in advance of the specific ADHRS mode/plant event evaluations following. These are treated below:

- ADHRS failures - Active failures in the ADHRS that may influence the outcome of an event when the ADHRS is operating include the following:

  -- Failure of ADHRS pumps to stop on demand (non-Q control/power circuitry)

  -- Failure of ADHRS pump low suction pressure trip to function on demand (non-Q control circuitry)

  -- Failure of ADHRS discharge valve to open or close on demand, and

-- Failure of ADHRS pump or discharge check valves to close on demand.

The first two items (above) are discussed here, and the third and fourth items are addressed in the specific mode/event evaluations following.

Failure of the ADHRS pumps to stop on demand may be due to a failure in the control circuitry or power supply breakers. Typically these events involve continued operation of the ADHRS with RHR 'C' in operation, and the ADHRS discharge isolation valve closes, without the ADHRS pump(s) stopping. In such an event, pump damage/failure may occur. This is not a safety concern relative to other systems functionality. However, long term operation at shutoff head may cause failure of the ADHRS pump motors. The probability that such a failure (already assuming that the operator has attempted to stop them from the main control room) would propagate past the pump breakers and into the plant non-Q (i.e., BOP) power supply system and cause other failures need not be assumed. This is based on the redundancy of protection provided in non-Q electrical circuits (including that of the ADHRS modification), and the concept that multiple failures of non-Q components in the absence of some extraneous mechanism need not be postulated when considered a safety-related event.

Failure of the ADHRS pumps to trip on low suction pressure is evaluated in same way as the above discussion. Failure may result in motor damage and shorting. However, propagation of this failure through multiple non-Q protection features (i.e., high current breaker trips) back to a safety-related interface (offsite power) need not be postulated.

Assessment of the parallel operation with the RHR 'C' train if the ADHRS pumps fail to trip is made in the various event/mode case evaluations following.

- Interaction with Safety- Related System Support Functions - The ADHRS has potential functional interactions with the RHR A, B and C trains. Relative to systems that support the operation of those systems, the following functions are identified:

  -- Power supply

  -- Indication

  -- Room cooling

  -- Pump seal and motor oil cooling

  -- Heat exchanger cooling

There is no interaction or interdependence between ADHRS power supplies or indication. The BOP power supply to the ADHRS pumps and discharge valve is isolated through several devices to RHR or other safety-related system power supplies.

There is no interaction between ADHRS indications and any other system.

The ADHRS does not employ existing room cooling capabilities for its operation. A separate room cooler, sized for ADHRS heat loads is used. The ADHRS pumps do not require seal or motor oil (i.e., bearing) cooling.

The ADHRS uses the PSW system for heat exchanger cooling, while RHR trains A and B use SSW. No functional interconnections between the two systems exist.

- Keep Full Capability - With the various ADHRS system alignments, either during standby or operation, the piping keep-full capability of interconnecting systems that are assumed to be operable for ECCS functions must be assured.

The normal keep-fill function for the RHR A, B and C loops is provided by jockey pumps that take suction downstream of each loop's suppression pool isolation valve (E12F004A, B and C). As seen in Table 2, jockey pumps A and B trip when their suction isolation valve (E12F082A or B) is not open, (which in turn close if valves E12F006A/B is open) and jockey pump 'C' has no interlock. The A and B jockey pumps discharge to their respective loops main pump discharge lines and minimum flow lines, while the 'C' jockey pump discharges to both the pump suction and discharge lines.

When the ADHRS operates via E12F006A or B and E12F066A or B, the corresponding suction valve, E12F004A or B, would be closed and the jockey pump would be tripped as a consequence of E12F006A or B being open. However, in this case, the ADHRS will provide the needed keep fill capability to the suction side, and through the pump and pump discharge check valve to the discharge side. The available pressure is static head on the ADHRS suction (as a minimum, that from the RPV with water level at the vessel flange) less line losses.

For loop 'C' when the ADHRS is operating in the flush mode, the 'C' piping is kept full by the ADHRS flow. For other modes when the ADHRS is discharging to the 'C' loop discharge line, valve F12F004C is closed and the jockey pump would be secured. In this case, the discharge side of loop C is maintained full by ADHRS flow. The suction side does not have an opportunity to drain since it is below the suppression pool level.

- Instrumentation Interaction - The ADHRS does not have any functional connections with pertinent instrumentation such as reactor level or temperature. No signals are provided to ADHRS components, and instrumentation added by

  ADHRS provides no signals to existing plant components except indirectly for the RHR 'C' pump minimum flow valve E12F064C. In this case, with ADHRS flow in the RHR 'C' discharge line above 1000 gpm, the valve open/close logic is affected (per Table 2, E12F064C opens when the C pump is running and flow is less than 1000 gpm, and closes when the pump is running and flow exceeds 1000 gpm). In this circumstance, if the 'C' pump were to start, the minimum flow valve would not open. This is of no consequence, however, as the only purpose in opening is to protect the RHR 'C' pump from damage due to inadequate flow, and in this case, a flow path for the 'C' pump already exists (i.e., valve E12F042C is open to the vessel for ADHRS flow).

  Based on the above there are no opportunities for interference or defeat of existing signals, or misleading indication to operators.

  The RHR 'C' jockey pump supplies the division II suppression pool level instrumentation for reference leg fill. As noted above, when ADHRS is in operation the jockey pump is secured. This fill function would be maintained when the ADHRS is running. However, per

Technical Specification in Tables 3.3.7.5-1 and 3.3.8-1, suppression pool level indication and associated actuation functions are not required to be operable in reactor modes 4 or 5. (To the extent that the level indication may be inoperable without this fill function, Technical Specification 3/4.5.3.c requires the verification of pool level by an alternate indicator at least once per 12 hours.)

For convenience, the remaining evaluation is divided into major cases that are a function of the assumed ADHRS mode and assumed initiating event. These cases are:

Case 1: ADHRS in vessel to vessel cooling mode (via either E12F006A/E12F066A or E12F006B/E12F066B) with a reactor high pressure event.

Case 2: ADHRS in flush mode with a reactor high pressure event.

Case 3: ADHRS in vessel to vessel cooling mode (via either E12F006A/E12F066A or E12F006B/E12F066B) with a LOCA.

Case 4: ADHRS fuel pool to vessel cooling mode with a LOCA.

Case 5: ADHRS in flush mode with a LOCA.

Case 6: ADHRS in vessel to vessel cooling mode (or backup) (via either E12F006A/E12F066A or Q1E12F006B/Q1E12F006B) with a loss of SDC.

Case 7: ADHRS in fuel pool cooling mode (or backup) with a loss of SDC.

Case 8: ADHRS in flush mode with loss of SDC.

Note that ADHRS in fuel pool to vessel cooling mode with a reactor high pressure event is not possible since the reactor head must be off and the cavity flooded for this mode in which case generation of high pressure is not possible.

To the extent that other variables apply, such operable ECCS combinations, cavity status, etc., these are treated as subcases in the following evaluations.

Case 1: ADHRS in Vessel to Vessel Cooling with Reactor High Pressure Event

Potential subcases are identified in Table 3. As there is no causal relationship between this event and a LOCA, operable ECCS combinations are not addressed. The origin of a high pressure event in cold shutdown could conceivably be either due to water addition, taking the vessel "solid" and increasing pressure, or due to loss of cooling, in which vessel heatup increases the pressure. The water addition case is considered too implausible to be postulated. Relevant to loss of cooling, this could be either due to loss of RHR SDC, with ADHRS in a backup mode, or loss of ADHRS with an RHR train in a backup mode (it is assumed the vessel is not solid in these circumstances). The ramifications of the loss of cooling are covered in cases 6 through 8.

A detailed evaluation of Case 1 events is provided in Table 4.

As indicated in Table 4, certain procedural steps are necessary to ensure that this category of events is acceptable.

Case 2: ADHRS in Flush Mode with Reactor High Pressure Event

Potential subcases are identified in Table 3. These cases are similar to Cases 1.5 and 1.6 in that with valves E12F066A and B closed, there is no potential exposure to reactor high pressure. With procedural requirements to have them closed, and to verify them closed on a reactor high pressure event, these cases are acceptable.

Case 3: ADHRS in Vessel to Vessel Cooling with a LOCA

The various subcases are delineated in Table 3. Only potential operable ECCS combinations pertinent to ADHRS interactions are included, e.g., combinations with HPCS are not listed. The nature of the LOCA is unspecified in this evaluation. As noted in Section III.C, the coolant loss would not be due to a pipe break or crack in the ADHRS or connected RHR systems, and it is presumed that the event is due to some form of inadvertent drainage occurrence or a "false" LOCA due to instrumentation malfunction. For conservatism, it is assumed that the drainage rate is fast enough so that operator intervention in advance of trip signals in the course of drainage does not occur.

As indicated in Cases 3.2 and 3.11 and 3.14, procedural requirements and policy decisions are required to avoid overpressurization of the ADHRS system and parallel operation of the ADHRS when RHR 'C' operates in the LPCI mode.

Case 4: ADHRS in Spent Fuel Pool to Vessel Cooling Mode with a LOCA

Subcases are listed in Table 3 and evaluated in Table 6. This mode is primarily intended to be used when maintenance or testing is being performed on the shutdown cooling common suction valves, E12F008 and E12F009. The first three subcases evaluated reflect this situation. Three other subcases assume RHR A or B (depending on which is the designated operable ECCS train) may be in shutdown cooling or suppression pool cooling/test modes. Other combinations of multiple functional loops and allowed modes have not been evaluated due to the high unlikelihood that they would need to be employed. Cases involving the RHR 'A' or 'B' loops are acceptable. Case 3.3, involving the RHR 'C' loop as the functional ECCS train, is acceptable assuming overpressurization and parallel flow conditions of ADHRS are accommodated as discussed in Case 3 above.

Case 5: ADHRS in Flush Mode with a LOCA

Various subcases are indicated in Table 3 and evaluated in Table 7. As seen in Table 7 all cases involving RHR A or B as the operable ECCS are acceptable. Cases involving RHR 'C' as the operable ECCS loop are unacceptable due to code boundary and parallel flow considerations. Certain identified subcases are highly unlikely (e.g., cavity flooded with RHR A or B in SDC mode, but RHR 'C' designated as the operable ECCS loop) but were included for completeness.

Case 6: ADHRS in Vessel to Vessel Cooling Mode with Loss of SDC

This case is viewed both from establishing ADHRS if originally in Standby, and establishing RHR SDC if ADHRS was originally in operation, as indicated in the subcases shown in Table 3. No specific reasons for the loss of an operating RHR SDC loop are postulated. However, in accordance with II.D.2.a, "hot short" variety of failures wherein a valve changes position without a demand are not postulated. Neither are system isolations due to a false LOCA signal or operator errors that change valve positions or stop pumps since these are recoverable with the RHR SDC loop that was operating. A "valid" failure would be an RHR pump motor failure.

For cases where ADHRS is in operation and is assumed to fail, the ADHRS is postulated to be operating via RHR 'A' or 'B' loop suction paths regardless of the operable RHR SDC backup train assumed. Typically, it is anticipated that ADHRS would operate using a suction path through the operable SDC train. However, as indicated in II.D.2.a, no pre-assumption is made as seen in the table.

As indicated in Table 8, all cases are acceptable, based on considerations that the ADHRS is equally reliable as a backup SDC source as an RHR train, as detailed in Table 8.

Case 7: ADHRS in Spent Fuel Pool to Vessel Cooling Mode with Loss of SDC

Subcases are shown in Table 3. Cases involving ADHRS as a backup cooling method to an operating RHR SDC train are not evaluated since only one RHR SDC train is required to be operable with the cavity flooded and an alternate cooling capability in this circumstance is not required per the Technical Specifications.

Evaluations are described in Table 9, and, as indicated there, the subcases are acceptable.

Case 8: ADHRS in Flush Mode with Loss of SDC

In this mode, ADHRS is not yet "qualified" as a backup shutdown cooling means, and Technical Specification requirements for operable RHR SDC loops apply. As the flush mode would only involve the RHR 'C' loop, and there are no interconnections with the 'A' or 'B' SDC trains in this mode (E12F066A and B and G41F059 are all closed) the ADHRS backup function or flush mode operation is not relevant to a postulated loss of an SDC train. Consequently, no subcases are indicated in Table 3, or evaluated.

III. PHYSICAL INTERACTIONS

A. AMBIENT CONDITIONS

Ambient conditions potentially affected by the ADHRS include:

- o Temperature
- o Radiation
- o Materials compatibility
- o Reactor water chemistry

The ADHRS would have no impact on other environmental parameters such as humidity or pressure.

1. Temperature

Operation of ADHRS will introduce heat loads in areas not previously designed for them. In the RHR 'C' pump room, an air-handling unit (AHU) is being added as part of the ADHRS modification to accommodate the additional heat load. The sizing basis of the AHU is to maintain the current normal condition (non-accident) room temperature below that assumed for environmental qualification of 105°F (per UFSAR Table 3.11-1).

The designation of the AHU as non-safety-related is consistent with the existing normal condition ventilation supply to the RHR 'C' pump room. Failure of this AHU is considered no more probable than the existing cooling system (loss of ventilation in this area is not considered in UFSAR Section 3.11.4).

For piping outside of the 'C' pump room, which is mainly existing piping, higher operating temperatures may exist in some lines than is currently the case (200°F rather than 140°F), which could increase temperatures in the various areas this piping is routed. However, this piping will be wrapped with lead for radiation shielding and with insulation for personnel protection, when required, to achieve 140°F outside surface temperature. As a result, temperature effects are expected to be minimal, particularly considered the unlikelihood that ADHRS would be operating at these elevated temperatures (i.e., mode 4 during ADHRS testing and if an operating RHR SDC loop fails). Consequently, no system recalculations or rebalancing is required.

2. Radiation

The ADHRS operates with radioactively contaminated water and poses the potential for increased doses in the areas where its piping is routed and equipment located.

For ADHRS piping outside the RHR 'C' and LPCS pump rooms, lead wrap shielding will be added to maintain the existing radiation zoning (Zone B) to allow continued general access and environmental qualification parameters.

ADHRS equipment and piping inside the RHR 'C' pump room and ADHRS piping in the LPCS pump room is not shielded, and the radiation zoning will increase to radiation Zone D at 56.6 mr/hr during the first day of potential operation (24 hours after shutdown), and decline to Zone C levels within 7 days and Zone B levels after the reactor cavity/upper containment pool is flooded and the reactor coolant is diluted. Maximum rates in the forward part of the room, where the RHR 'C' pump and related equipment is located is 8.3 mr/hr (Rad Zone C), decaying to Zone B (0.5 to 2.5 mr/hr) within 7 days. The additional integrated dose over 40 years is about 225 rads. Per UFSAR Table 3.11-2, the current integrated dose (normal plus accident) is about 5x10E4 rads, so that this additional dose will have little effect. Also, per UFSAR Table 3.11-2, system operating dose rate is 30 rads/hr, which is less than the maximum noted above but greater than the average that would be seen during ADHRS operation. Environmental qualification documentation has been reviewed and revised as necessary. No adverse impact in the environmental qualification of equipment in these rooms was noted.

The ADHRS is not designed for post-accident operation, and post-accident shielding and dose effects are therefore not addressed.

3. Materials Compatibility

The ADHRS is constructed primarily of carbon steel components, as are the existing systems to which it is connected (RHR and PSW). The ADHRS pumps and heat exchangers were originally designed for spent fuel pool cooling and thus are compatible with reactor coolant. No adverse interactions are identified due to material incompatibility.

4. Reactor Water Chemistry

The ADHRS provides no treatment to the reactor coolant and contains no materials that would interact adversely with or contaminate the nuclear boiler system.

When secured during non-outage periods, the system will be filled with demineralized water consistent with UFSAR commitments in Section 3.11.5.3 for the RHR system. The ADHRS will also be flushed prior to use, as is currently the case with the RHR system, as noted in UFSAR 3.11.5.3.

B. <u>IMPOSED LOADINGS</u>

The ADHRS design considers the loadings it imposes on existing systems to which it is connected or by which it is supported, and loading imposed on it by connected system. These loadings include the following:

- o Normal Loads (dead weight, pressure and thermal) - The existing plant piping that is to be used as part of the ADHRS reactor coolant flow path is requalified for higher design temperatures as necessary. Lead wrap shielding is to be added to some of this piping, and reanalysis is made for the additional weight.

- o Transient Loads - Waterhammer loads are to be avoided by operator procedures in starting the ADHRS, and are not included in the design. Adequate vents and drains are included in the design to provide for complete filling and venting prior to pump start.

- Seismic Loads - The ADHRS is de. ned for SSE loads.
- Hydrodynamic Loads - To the extent to which they may apply (i.e., transmission from the RHR 'C' discharge piping) they are included.
- Fatique - Increased usage of the ASME Class 1 portion of RHR 'C' discharge line requires a re-evaluation of fatigue usage. This has been performed by GE for the anticipated ADHRS maximum flow rate and found to be acceptable. Increased usage of existing non-ASME Class 1 piping by the ADHRS will not increase the number of cycles to the extent that fatigue evaluation is required.
- Loads on Supports and Structures - As imposed by ADHRS components, these are evaluated in the design.

C. HAZARDS CONDITIONS

1. Equipment to be Considered

The following additional equipment is added in the Auxiliary Building as a result of the addition of the ADHRS:

Equipment	Location and Data
New ADHRS pumps	RHR C pump room
New heat exchangers	RHR C pump room
Suction piping to pumps	Not in service during normal plant operation. RHR C pump room
Discharge piping	Not in service during normal plant operation. RHR C pump room
PSW supply to new heat exchangers and AHU	RHR C pump room and corridor outside RHR C pump room Not in service during normal plant operation.
PSW return from new heat exchanger and AHU	RHR C pump room and corridor outside RHR C pump room

Equipment	Not in service during normal plant operation. Location and Data
New AHU	RHR C pump room
Instruments	RHR C pump room Corridor out side RHR C pump room
MCC	Located in Auxiliary Building outside RHR 'C'
Cables (power)	Located in Auxiliary Building
Cables (controls)	Located in Auxiliary Building and Control Room

Previous analyses in the UFSAR have not considered effects of these components on safe shutdown capability of the plant.

This section addresses the effects of the hazards that could be potentially generated by this additional equipment.

2. Hazards Evaluation

The additional electrical connections to the pump motors and other loads are obtained from the existing MCCs. Loss of these MCCs due to other hazards analyzed (e.g., spray, internal flooding) in the area where the MCCs are located would cause loss of power to the ADHRS pumps. This will not prevent mitigation of the hazard causing such loss and bringing the plant to cold shutdown or maintaining in a safe condition since this equipment is not required for these or any other mitigating actions. The same rationale is also applicable to the new instruments and controls added by the ADHRS a and they are not considered further. Single failure criteria fulfillment, ramifications of loss of ADHRS and logic interactions are discussed in other sections of this report.

The potential hazards considered are:

a. High Energy Line Break (HELB)

b. Moderate Energy Line Break (MELB)

c. Internal flooding

d. Spray effects

e. Internally generated missiles

f. External hazards

g. Seismic hazards

h. Fire

Each item is discussed in this section.

3. Pipe Breaks

UFSAR section 3.6A.1.1 provides design bases that include Criteria, objectives and assumptions used in determining the postulated piping failures.

UFSAR Section 3.6A.1.1a states that this criteria conforms to Appendix A of 10 CFR 50, General design criterion 4, Environmental and Missile Design Bases, and also the overall design for this protection is in compliance with USNRC Regulatory Guide 1.46 and NRC Branch Technical positions (BTP) APCSB 3-1, and MEB 3-1. This criteria is applied to the new piping also.

a. High Energy Line Break (HELB)

UFSAR Section 3.6A.2 defines high energy piping as those systems or portions of systems in which the maximum operating temperature exceeds 200 F or the maximum operating pressure exceeds 275 psig during normal plant operating conditions.

The pressure and temperature in the new ADHRS system suction and discharge lines does not exceed these limits. The discharge from the ADHRS system is connected to the RHR C pump discharge line in the RHR C pump room. The discharge pressure of the RHR C pump is higher than the discharge pressure of ADHRS pumps. Hence the portion of ADHRS piping that is subjected to RHR C pump discharge pressure is designed to the same design pressure of RHR C pump discharge up to the isolation valves. This portion of pipe may be subjected to higher than 275 psig. However piping which exceeds 200 F or 275 psig for 2% or less of the time the system is in operation (or if the system was exposed to pressures or temperatures higher than the above limits for less than 1% of the plant operation) is considered as moderate-energy piping per the above criteria. In accordance with this definition, the UFSAR classifies the suction piping of RHR C and discharge piping from pump discharge to LPCI injection valve E12F041C as moderate-energy piping.

In accordance with the same definition, all the new fluid systems added as a result of this design are classified as moderate-energy piping. No further HELB analysis (jet impingement, pipe whip, etc.) is required.

b. Moderate-Energy Line Break (MELB)

UFSAR Section 3.6A.1.1.c.1 states that pipe breaks or cracks are postulated to occur during normal plant operation (i.e., reactor startup, operation at power, hot standby or reactor cooldown to cold shutdown). The ADHRS does not operate during these modes of normal plant operation. During these periods the system is shut down and isolated with the pump motor breakers racked-out. Thus, during the times a MELB must be postulated, the ADHRS and PSW systems hold dead volumes of water with no energy source.

A moderate energy crack is postulated on each line and evaluated for wetting from spray, flooding, and other environmental effects. A discussion of these evaluations follows.

c. Internal Flooding.

The new piping added is located in the RHR C pump room or in the corridor outside the RHR C pump room. If there is a moderate-energy crack, flooding would result in these areas. Such cracks are analyzed as necessary assuming the leakage area to be equal to the product of one-half the pipe wall thickness and-one half the pipe inside diameter, but circular in shape. Resulting flow rates are approximated using Crane Technical Paper 410 (1957), equation 2.14, with a flow coefficient of 0.6 and normal operating pressure for the pipe.

- RHR C Pump Room

  The UFSAR considered a moderate energy crack in the 24" RHR C pump suction line, with a pressure of 20 psig in the suction line. In accordance with the above criteria a flow of 329 gpm was calculated for a flooding rate of 42.2 minutes per foot. All components in the room are associated with RHR C or ADHRS except for one of the four suppression pool level monitors. Electrical circuits are division 2. During this event all the components in this room are postulated to be inoperable and the results were found to be acceptable in UFSAR Section 3C.4.2.6. This analysis did not include loss of ADHRS components located in the RHR C pump room. Addition of these components does not alter the results of UFSAR since ADHRS components are not required to mitigate the consequences of the crack postulated nor is ADHRS required to bring the plant to safe shut down.

Further postulating a crack in the ADHRS lines in the RHR C room (including PSW lines) would result in loss of a dead volume of water of only a few hundred gallons of water at most. The results of this event are clearly enveloped by to the results identified in the UFSAR for the RHR 'C' pipe crack.

In accordance with the criteria in Section 3.6.2 of UFSAR, pipe cracks were not assumed during Plant Conditions 4 or 5.

- o Corridor Outside RHR C Pump Room

New piping added in the main corridors are the PSW supply and return lines for the ADHRS system. These lines are connected to the main PSW lines in the corridors. UFSAR addresses the potential for flooding the main corridors in which the largest postulated leakage was from a 36-inch plant service water system line, with a maximum leakage of 668 gpm. The PSW lines added are 12-inch lines...smaller than 36-inch lines used in the analysis. Thus the case analyzed in the UFSAR will envelope the conditions.

d. Effects of spray

The approach taken to evaluate the effects of spray from moderate energy cracks is identical to flooding analysis.

- o Spray in RHR C Pump Room

From postulated cracks, all equipment in the room is considered inoperable. This renders RHR C and one of four suppression pool level monitors and ADHRS inoperable. Three other suppression pool level monitors would still be available. A crack on ADHRS line during normal plant operation causes trickling down of the dead volume rather than a spray since there is no energy source to sustain a spray. Employing the same rationale as in moderate-energy pipe failure evaluation above, adding ADHRS system components does not alter the results of the spray analysis included in the UFSAR.

o Spray in the Corridor Outside RHR C Room

UFSAR Section 3C.3.2 discusses the effects of spray in each compartment or significant area where essential equipment is located. This discussion did not include the auxiliary building corridors. This is due to the fact that there is no essential equipment located in the corridors. Potential spray from the the new PSW lines in the corridors is not considered as a concern.

e. Leakage Detection

Existing leakage detection in the RHR 'C' pump room consists of a sump high level alarm. This is considered adequate for the ADHRS. Although the RHR A and B trains have additional leakage detection related to the shutdown cooling mode (high area temperature and high ventilation differential temperature which isolate shutdown cooling suction valves E12F008 and E12F009 as indicated in Table 2) this is not considered necessary for ADHRS. This is because leakage due to a pipe break/crack is not postulated during reactor modes 4 or 5, as noted earlier. Consistent with this, the leakage detection/isolation circuitry for the RHR A and B loops is not required to be operable during modes 4 and 5, as seen in Technical Specification Table 3.3.2-1.

4. Internally generated missiles

There are two types of internally generated missiles that were considered in UFSAR:

a. Rotating Component Failure

The ADHRS pumps, the new AHU fan and the pumps in the radiation monitoring system are the only major components that could cause internally generated missiles that fall in the rotating equipment failure category. Section 3.5 of the UFSAR states that credible internally generated missiles are postulated only on rotating equipment that are likely to be in operation during normal plant operating conditions. Table 3.5.1 of the UFSAR lists such potential missile sources, and does not address RHR Pumps A, B or C. It follows, then, that it is not necessary to postulate internally generated missiles from the new equipment added by this design. (ADHRS pumps, the new AHU fan or the radiation monitoring system pumps are not operating during normal plant operating modes.)

b. Pressurized Component Failure

Missiles from a pressurized components failure are postulated only in case of High Energy Lines. Since the lines added by the ADHRS are moderate-energy, this is not a concern.

c. Turbine Missiles

The turbine missile analysis in the UFSAR is unaffected by the addition of the new ADHRS equipment as the ADHRS is not required for safe shutdown of the plant. During the times when turbine missile generation could occur (reactor mode 1) the ADHRS is secured and isolated such that damage to it would have no adverse consequential effects.

5. External Hazards

These include flooding, high winds, tornadoes, and tornado missiles.

External flooding is not a concern since the auxiliary building is designed with adequate watertight doors, hatches, and water stops. As the ADHRS is completely housed within existing safety-related structures, the same protection is offered as for existing safety-related components.

6. Seismic Hazards

ADHRS components are designed to withstand SSE loads, including non-safety-related components. Thus no hazard can occur to existing plant safety-related components from ADHRS components that fail during an earthquake. Additionally, safety-related ADHRS components are not susceptible to damage because all other components in rooms with ADHRS equipment are seismically supported.

7. Fire

The ADHRS does not introduce any significant new combustible loadings, and thus does not pose any significant fire hazard. Existing fire detection and suppression features in the RHR 'C' pump room are considered adequate for the added equipment.

Fire damage to the ADHRS does not pose any safety-related concern. To the extent that fire may damage the motor operators or power and control circuitry for valves E12F066A and B, no coincident or consequential event is postulated that requires the use (i.e., closure) of these valves.

D. PROCESS CONDITIONS

Process conditions of interest include:

- Flow - Sizing of added ADHRS piping and equipment has been reviewed for acceptable flow velocity. Velocities in existing plant piping used by the ADHRS have been reviewed and found acceptable.

- Pressure - ADHRS discharge pressure is lower than the design pressure of the RHR 'C' piping so that overpressure of existing piping cannot occur.

- Temperature - ADHRS process temperatures have been reviewed against the design temperatures of existing piping used by the ADHRS. As a result, the design temperature in some of the piping used in the suction path has been increased. The piping stress loads have been reanalyzed and were found to be acceptable.

E. OFFSITE EFFECTS

The direct radiation effects from operation of the ADHRS are negligible and are bounded by the effects of operating existing systems, such as RHR operation in Shutdown Cooling mode.

The ADHRS heat exchangers present a reactor water interface with the Plant Service Water (PSW) system in much the same manner as the existing reactor water interfaces with the Standby Service Water (SSW) system (RHR heat exchangers) and the Component Cooling Water (CCW) system (RWCU nonregenerative heat exchanger).

The potential for reactor coolant leakage to the environment was assessed in a conservative, bounding analysis which assumed that the entire reactor coolant volume at the Technical Specification reactor coolant activity limit (decayed for 24 hours) was released. (This assumption is equivalent to postulating the highly unlikely gross structural failure of the ASME III, Class 3, Seismic Category I ADHRS heat exchanger pressure boundaries.) The analysis further assumed that 10 percent of the iodine in the reactor coolant was immediately transported, unfiltered, to the site boundary using the 0-2 hour atmospheric dispersion factor.

The resultant offsite boundary doses are 1.19 rem thyroid and 0.00137 rem whole body.

These doses are significantly less than a small fraction of the 10CFR100 limits applicable for this event (10% x 300 rem = 30 rem thyroid; 10% x 25 rem = 2.5 rem whole body).

The GGNS UFSAR Section 15.7.2 addresses the consequences of a postulated unexpected and uncontrolled release of radioactivity due to a radioactive liquid waste system failure. The results of the assessment provided in Table 15.7-7 are a 1.25 rem thyroid dose and a "negligible" whole body dose. Considering a whole body dose of 0.00137 rem to be negligible, this assessment bounds the ADHRS heat exchanger rupture.

The limits in Regulatory Guide 1.26 Rev. 3 and 1.29 Rev. 3 for non-seismically designed equipment containing radioactive materials are 0.5 rem whole body or the equivalent to an organ, which is 3.0 rem for the thyroid. Although the ADHRS heat exchangers are seismically designed, the calculated doses from a postulated failure of this equipment are nevertheless below the guidelines for non-seismically designed equipment.

A radiation monitor is provided on the PSW common discharge line from the ADHRS heat exchangers to detect tube failures and leakage of reactor coolant into the PSW system. Since this monitor is not required to prevent exceeding safety limits, the monitor is not designated as safety related or powered from a Class 1E ESF source. A high radiation alarm is provided in the main Control Room to alert operators to an abnormal condition so that system isolation can be accomplished, if required. A low radiation alarm is also provided to indicate monitor failure. In that event, periodic grab samples can be taken to allow continued system operation.

Direct radiation effects from the operation of the ADHRS are negligible and are bounded by those that may occur from existing system operation, such as the RHR system in the Shutdown Cooling mode.

IV. CONCLUSIONS

Based on the material provided in Sections II and III, the ADHRS design is considered acceptable from an interactions point of view, assuming the procedural and other actions listed below are implemented:

- o The ADHRS should not be taken out of its isolated condition prior to entry into reactor mode 4 and should be isolated any time thereafter that exit from mode 4 to mode 3 or 2 occurs (II.A.1, Table 4).

- o Simultaneous operation of ADHRS and other systems found not acceptable in Section II.B will be precluded by procedure administrative controls.

- o RHR ECCS trains not assumed operable in the various cases evaluated in Section II.D should not be operated.

- o Protection from inadvertent drainage as identified in Table 1 should be invoked by procedure. Certain specific procedural requirements noted in Table 1 include:

  - Closure of valve E12F004A when operating in the vessel-to-vessel cooling mode via E12F006B/E12F066B and E12F004B when operating via E12F006A/E12F066A. (Cases 5 and 6)

  - Closure of E12F004C when operating in the vessel-to-vessel or spent fuel pool-to-vessel modes. (Cases 8 and 9)

  - Separate procedural steps to ensure E12F021C is closed. (Case 3)

- Verify valve E12F064C closed prior to ADHRS operation.

- Invoke existing inadvertent drainage provisions during SDC for RHR loops A or B when operating ADHRS via E12F066A or B, respectively.

o Procedural steps to preclude high pressure on the ADHRS suction and/or entry in mode 3 from mode 4 (See Table 4, Case 1.1 and Table 8, Case 6.1).

o Procedural steps to preclude high pressure on the ADHRS discharge and parallel flow with ADHRS when the RHR 'C' pump is operating, as indicated in Table 5, Cases 3.2, 3.11, 3.14 and Table 6.

o Prohibition of RHR 'C' as an operable ECCS train when ADHRS is in the flush mode (Table 7, Case 5.2).

o Adequate operating margin is provided to place ADHRS in operation to prevent exit from mode 4 into mode 3 (Table 8, Case 6.1).

TABLE 1

ADHRS - INADVERTENT VESSEL DRAIN EVALUATION

CASE NO.	DRAIN PATH DESCRIPTION	DRAIN PATH VIA	ACCEPTABILITY/RESOLUTION	REFERENCES
1	RPV to Suppression Pool	ADHRS suction path to RHR "C" suction: E12F009 and 8 (open) E12F006A or B (open) E12F066A or B (open) G41F059 (open) G41F057 (closed) E12F066C (locked closed) E12F004C to pool	Acceptable - requires 2 normally closed valves to be open, implies 2 operator errors.	P&ID M-1085A, Rev. 42 (RHR A) M-1085B, Rev. 36 (RHR B) M-1085C, Rev. 4 (RHR C) M-1088E, Rev. 1
2	RPV to Suppression Pool	ADHRS suction path to LPCS suction: E12F009 and 8 (open) E12F006A or B (open) E12F066A or B (open) G41F059 (open) G41F058 (closed) E21F036 (locked closed) E21F001A to pool	Acceptable - requires 2 normally closed valves to be open, implies 2 operator errors.	P&ID RHR P&IDs above M-1087, Rev. 23
3	RPV to Suppression Pool	ADHRS suction path to RHR A or B suction: E12F009 and 8 (open) E12F006A (open) E12F004A to pool	Acceptable - valves E12F006A and E12F004A are interlocked so that both cannot be open at the same time. Path would require 1 operator error (misposition of E12F004A) and failure of interlock.	P&ID RHR P&IDs - see above Schemes E-1191-001, Rev. 4 for E12F004A E-1181-001, Rev. 4 for E12F006A
4	RPV to Suppression Pool	Same as Case 3 above except via B loop valves.	Same as above except for B loop valves.	See 3 above.
5	RPV to Suppression Pool	ADHRS suction path to RHR A suction: E12F009 and 8 (open) E12F006B (open) E12F066A (closed) E12F004A to pool	No acceptable - 1 operator error (misposition of E12F066A) creates drain path. Resolution - Procedurally require E12F004A to be closed in this lineup or tag open breaker for valve E12F066A. 2 operator errors will then be required to create drain path. (Note: RHR Pump A is interlocked with E12F004A so pump will not start when valve closed.)	P&ID RHR P&IDs - see above Scheme E-1181-43, Rev. 8 for pump interlock. E-1181-005, Rev. 4 for E12F006B

TABLE 1

ADHRS - INADVERTENT VESSEL DRAIN EVALUATION

CASE NO.	DRAIN PATH DESCRIPTION	DRAIN PATH VIA	ACCEPTABILITY/RESOLUTION	REFERENCES
6	RPV to Suppression Pool	Same as Case 5 above except to RHR B suction.	Same as above except for B loop valve.	RHR P&IDs - see above. Scheme E-1181-44, Rev. 5 for pump interlock.
7	Spent Fuel Pool to Suppression Pool	ADHRS fuel pool suction path: G41F226 (open) G41F057 (closed) E12F066C (closed) E12F004C to pool or: G41F059 (closed) E12F066A or B (closed) E12F004A or B to pool	Acceptable - requires 2 normally closed valves to be open, implies 2 operator errors.	RHR P&IDs - see above.
8	RPV to Suppression Pool	ADHRS discharge path to RHR C (18"-GBB-58) suction: E12F029C (locked open) E12F031C (closed) E12F004C to pool	Not acceptable - 1 equipment failure (stuck open check valve E12F031) creates drain path. Resolution - Procedurally require E12F004C to be closed. 1 failure and 1 operator error (misposition of E12F004C) will then be required to create drain path. (Add interlock between E12F004C and RHR C pump to prevent pump start if E12F004C is closed to avoid pump damage.)	RHR P&IDs - see above.
9	RPV to Suppression Pool	ADHRS discharge path to RHR C (18"-GBB-58) suction: E12F029C (locked open) E12F022C (locked closed) E12F004C to pool	Not acceptable - 1 operator error (E12F022 mispositioned to open position) creates drain path. Resolution - same as Case 8 above.	RHR P&IDs - see above.
10	RPV to Suppression Pool	ADHRS discharge path to RHR C (18"-GBB-58) to RHR jockey pump: E12F029C (locked open) E12F085C (closed) E12F084C (closed) E12F082C (locked open) E12F004C to pool	Acceptable: 1) 2 failures required - stuck open check valves E12F085C and E12F084C 2) Part of path is 1"-GBB-67	RHR C P&ID - see above.

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TABLE1

TABLE 1

ADHRS - INADVERTENT VESSEL DRAIN EVALUATION

CASE NO.	DRAIN PATH DESCRIPTION	DRAIN PATH VIA	ACCEPTABILITY/RESOLUTION	REFERENCES
11	RPV to Suppresion Pool	ADHRS discharge path to RHR C (18"-GBB-58) to RHR jockey pump: E12F029C (locked open) E12F085C (closed) E12F084C (closed) E12F278 (open) E12F004C-B to pool	Acceptable : 1) 2 failures required - stuck open check valves E12F085C and E12F084C 2) Parts of path are 1" (GBB-67 and GBB-68)	RHR C P&ID - see above.
12	RPV to Suppression Pool	ADHRS discharge to RHR C 18"-GBB-59 to minimum flow line (4"-HBB-115): E12F029 (locked open) E12F022 (locked closed) or E12F031C (closed) E12F064C (open) E12F046 (open) E12F018C (locked open) to pool	Not acceptable - Failure of E12F031C (stuck open) or improper alignment (E12F022 open) will result in drainage through minimum flow bypass valve F064C to suppression pool. Resolution: Require F064C to be closed when ASDCS is in operation.	P&IDs RHR - see above M-1077B, Rev. 21 M-1077C, Rev. 25 Schemes E-1181-18, Rev. 4 (F064C) E-1181-45, Rev. 6 (C pump) E-1160-9, Rev. 4 (F008) E-1160-10, Rev. 6 (F00?)
13	RPV to Suppression Pool	ADHRS discharge to RHR C (4"-HBB-115) F021 (closed) to pool	Unacceptable - 1 operator error (misposition of E12F021) creates drainage path. Resolution - Require two separate procedural steps so that two separate operator errors are required: 1) ADHRS system lineup - verify E12F021C closed and MO breaker racked out. 2) Prior to ADHRS operation (opening of suction valves E12F066A or B) verify E12F021C closed. This can be done based on indication in the main control room.	RHR P&IDs - see above. Scheme E-1181-23, Rev. 4 for E12F021
14	RPV to Suppression Pool	ADHRS discharge to RHR C (18"-GBB-58) to 12"-GBB-131 P60FG21A (closed, fails closed) P60F001B (closed, fails closed) etc. to pool	Acceptable - At least 2 failures or 2 operator errors (misposition) required to create drain path.	RHR P&IDs - see above. M-1099, Rev. 10

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TABLE 1

ADHRS - INADVERTENT VESSEL DRAIN EVALUATION

CASE NO.	DRAIN PATH DESCRIPTION	DRAIN PATH VIA	ACCEPTABILITY/RESOLUTION	REFERENCES
15	RPV to Condensate and Refueling System	ADHRS discharge to RHR C to CS and RFW storage and transfer line (4"-GBB-101) E12F063C (locked closed) P11F007 (open) to various locations or P11F424 (open) to various locations	Potentially unacceptable (dependent on alignments upstream of F007 F424) Resolution - procedurally require verification that E12F063C is locked closed prior to ADHRS operation.	
16	RPV to LRW Surge Tank	ADHRS discharge to RHR C (18"-GBB-58) E12F029C (locked open) E12F022C (locked closed) or E12F031C (closed) E12F072C (locked closed) E12F070B (locked closed) E12F203 to surge tank	Acceptable - 3 operator errors (valve mispositions) required to establish path.	
17	RPV to RHR Loop A or B to Various RHR A or B Discharge Points (including suppression pool)	ADHRS suction path: E12F009 and B (open) E12F006A or B (open) to RHR 'A' or 'B' discharge loop (various paths following)	Not acceptable - various potential paths to suppression pool (e.g., via F029A, F064A and F048A) Resolution - require same RHR lineup precautions and restrictions as used when RHR is in th shutdown cooling mode.	RHR P&IDs - see above.
18	RPV to Suppression Pool	ADHRS in flush mode: RPV via RHR A or B in SDC mode. E12F066A or B (closed) G41F059 (closed) to ADHRS suction E12F021 to pool	Acceptable - requires 2 normally closed valves to be open, implies 2 operator errors.	RHR P&IDs - see above. M1088E, Rev. 1

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TABLE 1

ADHRS - INADVERTENT VESSEL DRAIN EVALUATION

CASE NO.	DRAIN PATH DESCRIPTION	DRAIN PATH VIA	ACCEPTABILITY/RESOLUTION	REFERENCES
19	RPV to Suppression Pool	ADHRS in flush mode: RPV via RHR A or B in SDC mode. E12F066A or B (closed) G41F059 (closed) G41F057 (open) E12F066C (open) F004C (open) to pool	Acceptable - requires 2 normally closed valves to be open, implies 2 operator errors.	RHR P&IDs - see above. M1088E, Rev. 1
Other	Various	Various through 1" lines	Acceptable - 1" drain paths are not required to be considered per criteria.	

NOTE: Potential drainage cases due to simultaneous operation of ADHRS and RHR in modes other than standby or SDC (Cases 18 and 19) are evaluated in Section II.B of the evaluation.

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TABLE 2

POTENTIAL CONTROLS INTERFACE POINTS

Pump or Valve		Pump Start or Valve Open		Pump Stop or Valve Close		Alarm/Indication	Reference
Number	Function	Auto Note 1	Manual Note 2	Auto Note 1	Manual Note 2	Note 3	Drawing
E12C002A/B	RHR Pump A/B	LPCI CS	HS-M600A/B HS-M200A/B HS(Local)	A	HS-M600A HS-M200A HS(Local)	RX Level Low Dry Well Press High Containment Press High Pump Auto Start Pump Auto Stop (light only)	E-1181-43, Rev. 8 E-1181-44, Rev. 5
E12C002C	RHR C Pump	LPCI	HS-M600C HS-Local	N/A	HS-M600C HS-Local	RX Level Low Dry Well Press High Pump Auto Start Pump Auto Stop (light only)	E-1181-45, Rev. 6
E12C003A/B	Jockey Pump A/B	N/A	HS-M601A/B and F082A/B Full Open	F082A/B Not Fully Open	HS-M601A/B	Overload or Power Loss	E-1180-02, Rev. 8
E12C003C	Jockey Pump C	N/A	HS-M601C	N/A	HS-M601C	Overload or Power Loss	E-1181-46, Rev. 3
E12F003A/B	Heat Exchanger Shell Discharge	N/A	HS-M607A/B HS-M207A/B	N/A	HS-M607A/B HS-M207A/B	Overload or Power Loss	E-1181-06, Rev. 6 E-1181-07, Rev. 5
E12F004A/B	Suppression Pool Suction	N/A	HS-M602A/B and F006A/B Closed. HS-M202A/B	N/A	HS-M206A HS-M202A	Overload or Power Loss	E-1181-1, Rev. 4 E-1181-2, Rev. 2
E12F004C	Suppression Pool Suction	N/A	HS-M602C HS-M202C	N/A	HS-M602C HS-M202C	Overload or Power Loss	E-1181-3, Rev. 4
E12F006A/B	Shutdown to A/B Loop Pump	N/A	HS-M605A/B and (F004A/B and F024A/B) Closed HS-M205A/B	N/A	HS-M605A/B HS-M205A	Overload or Power Loss	E-1181-4, Rev. 4 E-1181-5, Rev. 4
E12F008	Outboard Shutdown Valve	N/A	HS-M604 and NSDC	SDC	HS-M604 HS-M204	Overload and Power Loss Low RPV Level 3 RPV Pressure High RHR Equip Area Temp High RHR Equip Area Vent dt High	E-1160-9, Rev. 4
E12F009	Inboard Shutdown Valve	N/A	HS-M603 and NSDC	SDC	HS-M603 HS-M203	Overload and Power Loss Low RPV Level 3 RPV Pressure High RHR Equip Area Temp High RHR Equip Area Vent dt High	E-1160-10, Rev. 6

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TABLE 2

POTENTIAL CONTROLS INTERFACE POINTS

Pump or Valve		Pump Start or Valve Open		Pump Stop or Valve Close		Alarm/Indication	Reference
Number	Function	Auto Note 1	Manual Note 2	Auto Note 1	Manual Note 2	Note 3	Drawing
E12F011A/B	Condensate Discharge to Pool	N/A	HS-M631A/B HS-M231A/B	LPCI	HS-M631A/B HS-M231A/B	Overload or Power Loss	E-1181-08, Rev. 4 E-1181-09, Rev. 2
E12F021A/B	"C" Loop Discharge to Pool	N/A	HB-M622C and NLPCI	LPCI	HS-M622C	Overload or Power Loss	E-1181-23, Rev. 4
E12F023	RPV Head Spray Valve	N/A	HS-M632	SDC	HS-M632	Overload or Power Loss	E-1160-11, Rev. 6
E12F024A/B	A/B Loop Dis- to Pool	N/A	HS-M622A/B and NCS; HS-M222A/B	LPCI CS	HS-M622A/B HS-M222A/B	Overload or Power Loss	E-1181-10, Rev. 5 E-1181-11, Rev. 4
E12F026A/B	Condensate Discharge to RCIC	N/A	HS-M626A/B HS-M226A/B	LPCI	HS-M626A/B HS-M226A/B	Overload or Power Loss	E-1181-12, Rev. 3 E-1181-13, Rev. 2
E12F027A/B	A/B Loop Containment Isol.	LPCI	HS-M611A/B and AI; HS-M211A/B and AI	N/A	HS-M611A/B and NLPCI; HS-M211A/B and NLPCI	Overload or Power Loss	E-1181-39, Rev. 4 E-1181-40, Rev. 3
E12F028A	A Loop Containment Spray	CS	HS-M610A and F027A Closed. HS-M553 and F027A	LPCI and NCS	HS-M610A HS-M553	Overload or Power Loss	E-1181-41, Rev. 7
E12F028B	B Loop Containment Spray	CS	HS-M610B and F027A Closed.	LPCI and NCS	HS-610B	Overload or Power Loss	E-1181-42, Rev. 7
E12F037A/B	A/B Loop Return to Upper Pool	N/A	HS-M19A/B and NLPCI	LPCI	HS-619A/9	Overload or Power Loss	E-1181-24, Rev. 7 E-1181-25, Rev. 5
E12F040	Flush to Radwaste	N/A	HS-M636 and NB and NC; HS-M236	B, C	HS-M636 HS-M236	Overload or Power Loss	E-1160-12, Rev. 3

TABLE2

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TABLE 2

POTENTIAL CONTROLS INTERFACE POINTS

Pump or Valve		Pump Start or Valve Open		Pump Stop or Valve Close		Alarm/Indication	Reference
Number	Function	Auto Note 1	Manual Note 2	Auto Note 1	Manual Note 2	Note 3	Drawing
E12F042A/B	LPCI Injection Valve	LPCI and NCS	HS-M609A/B and (LPCI or BI) and NCS; HS-M209A/B and NCS	CS	HS-609A/B HS-209A/B	Overload or Power Loss Low Press Downstream of F042A/B	E-1181-37, Rev. 9 E-1181-38, Rev. 7
E12F042C	LPCI Injection Valve	LPCI	HS-M609C and (LPCI or BI)	N/A	HS-M609C	Overload or Power Loss	E-1181-036, Rev. 5
E12F047A/B	Heat Exchanger Shell Inlet	N/A	HS-M606A/B HS-M206A/B	N/A	HS-M606A/B HS-M206A/B	Overload or Power Loss	E-1181-14, Rev. 4 E-1181-15, Rev. 3
E12F048A/B	Heat Exchanger Shell Bypass	LPCI	HS-608A/B and NCS; HS-208A/B	CS	HS-M608A/B HS-M208A/B	Overload or Power Loss	E-1181-26, Rev. 4 E-1181-27, Rev. 2
E12F049	Flush to Radwaste	N/A	HS-M635 and NB and NC; HS-M235	B, C	HS-M635 HS-M235	Overload or Power Loss	E-1160-13, Rev. 3
E12F051A/B	Steam Regulating Valve	N/A	HS-629A/B	LPCI	HS-M629A/B	N/A	E-1181-73, Rev. 11 E-1181-75, Rev. 8
E12F052A/B	Steam Supply Isolation	N/A	HS-M627A/B and NLPCI HS-M227A/B	LPCI	HS-M627A HS-M627A	Overload or Power Loss	E-1181-16, Rev. 6 E-1181-17, Rev. 7
E12F053A/B	Shutdown Return to Feedwater	N/A	HS-M615A/B and NSDC; HS-M215A/B	SDC	HS-M615A/B HS-M215A/B	Overload or Power Loss	E-1181-28, Rev. 6 E-1181-29, Rev. 4
E12F060A/B	Water Sampling Valve	N/A	HS-M624A/B and NLPCI	LPCI	HS-M624A/B	Note 3	E-1160-52, Rev. 11
E12F064A/B/C	RHR Pump Min. Flow Control	RHR Pump Running Below Min. Flow	HS-M621A/B/C	Above Min. Flow	HS-M621A/B/C	Overload or Power Loss	E-1181-34, Rev. 5 E-1181-35, Rev. 4 E-1181-18, Rev. 4
E12F066A/B/C	RHR and ASDCS Pump Suction Valve	N/A	HS-(CR)	N/A	HS-(CR)	N/A	Figure 4.1

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TABLE2

TABLE 2

POTENTIAL CONTROLS INTERFACE POINTS

Pump or Valve Number	Pump or Valve Function	Pump Start or Valve Open Auto Note 1	Pump Start or Valve Open Manual Note 2	Pump Stop or Valve Close Auto Note 1	Pump Stop or Valve Close Manual Note 2	Alarm/Indication Note 3	Reference Drawing
E12F424	ASDCS Pump Discharge Valve	N/A	HS-(CR)	N/A	HS-(CR)	N/A	Figure 4.1
E12C005A/B	ASDCS Pump A/B	N/A	HS-(CR)	N/A	HS-(CR)	N/A	Figure 4.1
E21C001	LPCS Pump	LPCS	E21-HS-M610 E21-HS-(Local)	N/A	E21-HS-M610 E21-HS-(Local)	Pump Auto Start Pump Auto Stop (light only)	E-1182-06, Rev. A
E12F073A/B	Non-Condensable Vent from HX	N/A	HS-M634A/B	HS-M634A/B	N/A	Overload or Power Loss	E-1181-30, Rev. 4 E-1181-31, Rev. 3
E12F074A/B	Non-Condensable Vent from HX	N/A	HS-M633A/B	N/A	HS-M633A/B	Overload or Power Loss	E-1181-32, Rev. 4 E-1181-33, Rev. 5
E12F075A/B	Water Sampling Valve	N/A	HS-M625A/B and NLPCI	LPCI	HS-M625A/B	Note 3	E-1160-52, Rev. 11
E12F082A/B	RHR Jockey Pump Suction Valve	N/A	HS-M643A/B and E12F005A/B Full Closed	F006A/B Not Closed	HS-M643A/B	Overload or Power Loss	E-1180-07, Rev. 0 E-1180-03, Rev. 5
E12F087A/B	Low Pressure Steam Supply	N/A	HS-M628A/B and NLPCI and ND	LPCI D	HS-M628A/B	Overload or Power Loss	E-1181-19, Rev. 5 E-1181-20, Rev. 3
E12F094	Service Water to RHR Crosstie	N/A	HS-M640 HS-M240	N/A	HS-M640 HS-M240	Overload or Power Loss	E-1181-21, Rev. 3
E12F096	Service Water to RHR Crosstie	N/A	HS-M641 HS-M241	N/A	HS-M641 HS-M241	Overload or Power Loss	E-1181-22, Rev. 4
Reactor Pressure Vessel (RPV)						Low RPV Level 3 Low RPV Level 2 Low RPV Level 1 RPV Pressure High (135 psig)	
Drywell						Drywell Pressure High	
RHR Equip Area						RHR Equip Area Temp High RHR Equip Area Vent At High	

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NOTES

1. Auto signal codes are as follows:

   LPCI: Reactor vessel low water level 1 (-150") and/or high drywell pressure (1.39 psig). In a logic one-out-of-two taken twice (that is, two levels, or two pressure, or one level and one pressure).

   CS: High drywell pressure (1.39 psig) and/or containment pressure (7.84 psig) in a logic one-out-of-two taken twice.

   LPCS: Reactor vessel low water level 1 (-150") and/or high drywell pressure (1.39 psig) in a logic one-out-of-two taken twice.

   SDC: Reactor vessel low water level 3 (11'-4"), reactor vessel pressure high (135 psig), RHR equip area temp high or RHR equip area vent high dt (99°F).

   A: (E12F008, E12F009 or E12F006A/B closed) and (E12F004A/B and E12F066A/B closed)

   B: Drywell pressure high (1.23 psig)

   C: Reactor vessel low water level 3 (11.4")

   D: Main steam pressure high

2. Interlock signal codes are as follows:

   NSDC: Reactor vessel level and pressure, RHR equip area temperature and RHR equip area vent dt satisfactory.

   NLPCI: Reactor vessel level and drywell pressure satisfactory.

   NCS: Drywell pressure and containment pressure satisfactory.

   AI: E12F042A/B and E12F028A/B closed.

   NB: Drywell pressure satisfactory.

   NC: Reactor vessel level satisfatory.

   BI: Low pressure (<50 psig) downstream of E12F042A/B.

   ND: Main steam pressure normal.

NOTES (Continuation)

3. Every pump in this table has a red light for pump running and a green light for pump not running in the main control room.

   Every electric operated valve has a red light for valve open and green light for valve closed in the main control room.

TABLE 3 SHEET 1 OF 2

CONTROLS/OPERATIONAL INTERACTIONS

EVALUATION

CASE NO.	REACTOR MODE 4	REACTOR MODE 5	ASDCS MODE RPV/RPV VIA A	ASDCS MODE RPV/RPV VIA B	ASDCS MODE FP/RPV	ASDCS MODE SP/SP	REACTOR CAVITY DRAINED	REACTOR CAVITY FLOODED	EVENT HP	EVENT LOCA	EVENT LOSS SDC	ECCS OPERABLE A+LPCS	ECCS OPERABLE B&C	ECCS OPERABLE A&B	ECCS OPERABLE A&C	SDC OPERABLE A	SDC OPERABLE B	SDC STANDBY	RHR STATUS SDC	RHR STATUS SPC	RHR STATUS FPA
1.1	X		X				X		X							X		X			
1.2	X		X				X		X								X	X			
1.3	X			X			X		X							X		X			
1.4	X			X			X		X								X	X			
1.5	X		STANDBY				X		X			ECCS NOT PERTINENT				X			X		
1.6	X		STANDBY				X		X								X		X		
2.1	X					X	X		X							X	X	A	B		
2.2	X					X	X		X							X	X	B	A		
3.1	X	X	X				X			X		X						X			
3.2	X	X	X				X			X			X					X			
3.3	X	X	X				X			X				X				X			
3.4	X	X	X				X			X					X			X			
3.5	X	X		X			X			X		X						X			
3.6	X	X		X			X			X			X					X			
3.7	X	X		X			X			X				X				X			
3.8	X	X		X			X			X					X			X			
3.9		X	X					X		X		A				SDC NOT PERTINENT		X			
3.10		X	X					X		X			B					X			
3.11		X	X					X		X					C			X			
3.12		X		X				X		X		A						X			
3.13		X		X				X		X			B					X			
3.14		X		X				X		X					C			X			
4.1		X			X			X		X		A						X			
4.2		X			X			X		X			B					X			
4.3		X			X			X		X					C			X			
4.4		X			X			X		X		A							X		
4.5		X			X			X		X			B						X		
4.6		X			X			X		X		A								X	
4.7		X			X			X		X			B							X	

TABLE 3 SHEET 2 OF 2

CONTROLS/OPERATIONAL INTERACTIONS

EVALUATION

CASE NO.	REACTOR MODE 4	REACTOR MODE 5	ASDCS MODE RPV/RPV VIA A	ASDCS MODE RPV/RPV VIA B	ASDCS MODE FP/RPV	SP/SP	REACTOR CAVITY DRAINED	REACTOR CAVITY FLOODED	HP	EVENT LOCA	LOSS SDC	ECCS OPERABLE A+LPCS	ECCS OPERABLE B&C	ECCS OPERABLE A&B	ECCS OPERABLE A&C	SDC OPERABLE A	SDC OPERABLE B	STANDBY	RHR STATUS SDC	RHR STATUS SPC	RHR STATUS FPA
5.1	X	X				X	X			X		X				X	X	X	X		
5.2	X	X				X	X			X			X			X	X	X	X		
5.3	X	X				X	X			X				X		X	X	X	X		
5.4	X	X				X	X			X					X	X	X	X	X		
5.5		X				X		X		X		A				X			X		
5.6		X				X		X		X			B				X		X		
5.7		X				X		X		X					C	X or	X		X		
5.8		X				X		X		X					C	X or	X		X		
6.1	X		X (STANDBY)				X				X					X			X		
6.2	X			X (STANDBY)			X				X						X		X		
6.3		X	X (STANDBY)				X or	X			X					X			X		
6.4		X		X (STANDBY)			X or	X			X	ECCS NOT PERTINENT					X		X		
6.5		X	X				X or	X			X					X		X			
6.6		X		X			X or	X			X						X	X			
6.7		X	X				X or	X			X						X	X			
6.8		X		X			X or	X			X					X		X			
6.9	X		X				X				X					X		X			
6.10	X			X			X				X						X	X			
7.1		X			X		X				X	ECCS NOT PERTINENT				X		X			
7.2		X			X		X				X						X	X			

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TABLE 4

CASE 1 EVALUATIONS
(ADHRS in Vessel-to-Vessel Cooling with
Reactor High Pressure Event)

1. Case 1.1 Evaluation:

Initial Conditions: ADHRS operating in vessel to vessel cooling in Mode 4 after postulated failure of RHR SDC.

E12F004A closed
E12F008 and E12F009 open
E12F006A open
E12F066A open
(See Table 3 for other information)

Sequence of Events:

1. ADHRS cooling function lost.

2. Vessel pressure increases due to temperature increase. (Vessel pressure indication available in main control room).

3. Operator remote manually isolates ADHRS suction path valve (E12F066A) upon loss of the ADHRS cooling function.

4. High/pressure low pressure interlocks shut valves E12F008 and E12F009 at 135 psig (if operable) if not already shut by operator, and if shutdown cooling not restored.

Evaluation:

Initiation of safety functions:

- No interference on HP/LP interlocks on E12F008 or E12F009

- No interference with remote manual isolation of ADHRS suction indicated. Adequate time available (see discussion below)

Inadvertent drainage: See Table 1 for initial lineup. Response to event does not create a new flow path for inadvertent drainage.

ADHRS pressure integrity preserved: Yes, with isolation of suction line.

Single failure/operator error: 4 different safety-related motor operated valves are available to isolate ADHRS suction, single failure acceptable. Operator error to not isolate ADHRS suction could cause overpressure of ADHRS suction if cooling is not restored, see discussion below.

Significant Equipment Damage: None indicated.

Discussion:

1. Based on the expected heat loads it is judged that adequate time is available for operator action to isolate the ADHRS suction before design pressures are exceeded.

2. The operator has reactor pressure and temperature and ADHRS heat exchanger discharge temperature, flow, and pump running status indication in the main control room. To preclude a single operator error from accomplishing the isolation function, more than one procedural requirement should be instituted:

   o Isolate ADHRS suction whenever reactor temperature exceeds 200 degrees. This is required in any event since exceeding 200 degrees puts the reactor in mode 3, which is not a design basis mode for ADHRS operation.

   o Isolate ADHRS suction in mode 4 whenever ADHRS cooling function is lost (i.e., pumps stop, etc.).

Conclusion:

This case is acceptable subject to the implementation of the procedural actions indicated in Discussion Item 2 above.

2. <u>Case 1.2 Evaluation</u>

This involves the same scenario as Case 1.1 above, except that the B RHR SDC train is assumed to be the operable train. As seen in the evaluation of Case 1.1, the sequence of events, the system response, and the various elements of the evaluation and discussion are not dependent on which RHR SDC train was in standby. Therefore, the conclusions remain the same.

3. <u>Case 1.3 and 1.4 Evaluation</u>

Initial Conditions: ADHRS operating in vessel-to-vessel cooling in Mode 4 after postulated failure of RHR SDC.

E12F004B closed
E12F008 and E12F009 open
E12F006B open
E12F066B open
(See Table 3 for other information)

These are the same as Cases 1.1 and 1.2 except that ADHRS is assumed to be operating via E12F006B and E12F066B. For the reasons cited in Case 1.2 above, the content of Case 1.1 evaluation is applicable to these cases.

4. Case 1.5 and 1.6 Evaluation

Initial Conditions: ADHRS is standby with either RHR A or B SDC in operation.

E12F008 and E12F009 open
E12F006A open, E12F004A closed
(Case 1.5) or
E12F006B open and E12F004B closed
(Case 1.6)
E12F066A and B closed

Sequence of Events:

1. RHR SDC cooling function lost

2. Vessel pressure increases due to temperature increase.

3. Operator remote manually isolates RHR SDC suction path (E12F008 and 9, E12F006A or B) before RHR SDC suction design pressure exceeded, or automatic isolation of E12F008 and 9 occurs at 135 psig (if interlock operable).

These reflect the normal anticipated RHR SDC/ADHRS mode combinations for reactor Mode 4 with an RHR SDC train in operation and ADHRS in standby.

Based on procedural requirements that E12F066A and B are shut when ADHRS is in standby and the considerations in Case 1.1 above that there are no relationships between high pressure detection circuitry or isolation functions and the ADHRS, these cases are evaluated as acceptable.

Note: Case 2 Evaluations are covered in Section II.D.2.g.3 of the Interaction Evaluation.

TABLE 5

CASE 3 EVALUATIONS
(ADHRS in Vessel-to-Vessel Cooling with a LOCA)

1. <u>Case 3.1 Evaluation</u>:

Initial Conditions: ADHRS operating in vessel to vessel cooling via E12F006A/E12F066A with postulated LOCA event

E12F004A closed
E12F008 and E12F009 open
E12F006A open
E12F066A open
E12F064A closed
E12F064C closed
E12F021 closed

Sequence of Events:

1. Vessel drainage commences.

2. Vessel level decreases to Level 3, E12F008 and E12F009 automatically close.

3. ADHRS pumps trip on low suction pressure.

4. Vessel level decreases to Level 1, RHR 'A' and LPCS pump receive auto initiation signal for vessel injection.

   Permissive for RHR 'A' pump prevents pump start as valve E12F004A is closed, and pump trip bypass switch for valve E12F[illegible]06A is armed.

5. Operator closes valve E12F006A (to clear interlock with valve E12F004A).

   Valve E12F066A closed to isolate ASME III/Class 3 portion of ADHRS. ADHRS secured.

   Valve E12F004A opened. RHR 'A' pump is started remote manually.

Evaluation:

Initiation [illegible] Safety Functions:

- No inhibitions of initiating isolation or pump activation functions are identified.

- RHR A LPCI flow path can be established:

  E12F004A can be opened remote manually.
  E12F048A opens automatically on LPCI signal.

E12F027A opens automatically on LPCI signal.
E12F064A opens automatically on RHR A pump start.
E12F042A opens automatically on LPCI signal.
Other parts are isolated (e.g., containment spray (E12F028A), return to upper pool (E12F037A), condensate discharge to pool (E12F011A), etc.). None of the above have any functional relationship to any ADHRS control function or valves involved in the ADHRS unique system lineup (e.g., E12F066A).

- o No LPCI flow bypasses are indicated. ADHRS interface with RHR A is on the suction side. Taking suction from the vessel via the ADHRS/RHR C flow path via valve E12F066A is precluded by 3 check valves (E12F041C, E12F416, and E12F412A or B) in series.

- o No interference with RHR A support functions is indicated - there are no interactions with pump cooling or room ventilation systems (including SSW supply) or electrical power sources.

Operability of ECCS while in Standby:

- o Keep fill function maintained as discussed in Section II.D.2.g.3.

- o Non-LPCI mode lineup (requiring remote manual realignment of valves E12F004A, E12F006A and E12F066A) acceptable per GGNS Technical Specifications 3/4.5.2.

- o Alignment of RHR A loop for ADHRS operation does not interfere with minimum flow function of RHR A.

Minimum flow: No interference with E12F064A function indicated on RHR 'A' pump startup.

Inadvertent Drainage:

- o See Table 1 for initial lineup.

- o Potential for drainage during incorrect system realignment discussed below under single failure/operator error.

ADHRS pressure integrity: No interface between high pressure source (i.e., RHR 'A' discharge pressure) and ADHRS exists.

Single Failure/Operator Error:

- o Failure of E12F066A to close (or be closed by the operator) - This defeats the establishment of an ASME III/Class 2 pressure boundary for ECCS but otherwise has no functional effect as the piping beyond E12F066A is safety-related, seismic Category I, and designed for the process conditions that would occur in the LPCI mode (the potential for backflow is discussed above). As the valve and its motor operator are safety-related, its failure forecloses the assumption that another failure occurs that would defeat the redundant LPCS system.

- o Failure of valve E12F004A or E12F006A to change position would defeat the LPCI function but would not imply any additional deleterious consequences due to ADHRS because of its lineup (i.e., position of E12F066A). Valve E12F066A should be closed in any event to ensure that ADHRS does not continue to operate (although there is no obvious adverse consequences if it did in this case). Further details are provided in Case 3.2 below.

- o If the E12F066A pump start permissive bypass switch failed or was not positioned properly, the RHR 'A' pump could start without a suction source if E12F008 and E12F009 were closed automatically on level 3. However, as this is a safety-related component, its failure or misposition (procedural error) would mean that a single failure in the redundant LPCS system need not be considered. In such a case the RHR 'A' pump may continue to run and be damaged, however, this is not a safety concern and, relative to a component failure, is no more probable that other failures in the permissive circuitry.

- o If the operator intervenes as level is decreasing and shuts E12F008, E12F009 or E12F006A but fails to open E12F004A and the E12F066A permissive bypass fails, the same lack of pump suction situation occurs as described above. However, this would involve an operator error and single failure, which need not be postulated.

- o If the operator intervenes and starts RHR Pump A before level 3 isolation of E12F008 and E12F009 occurs (if operable), and does not realign the suction for LPCI, suction would occur on the vessel. In this case some drainage of the reactor vessel to the suppression pool would occur through the minimum flow bypass line (valve E12F064A) until flow increases and E12F064A closes. This may occur regardless of the use of ADHRS. Continue pumpdown of the vessel would occur if E12F064C fails to close or if the operator fails to open the LPCI injection valve. However, this would involve a second single failure/operator error and need not be postulated.

Significant equipment damage: As indicated above, with multiple failures the RHR pump could start without a suction path which could result in damage to the pump if it is not stopped expeditiously.

Conclusion:

This case is acceptable.

2. Case 3.2 Evaluation

Initial conditions: ADHRS in vessel to vessel cooling via E12F006A/E12F066A with LOCA event

E12F004A closed
E12F008 and E12F009 open
E12F006A open
E12F066A open

E12F004B open
E12F004C closed
E12F064A Closed
E12F006B closed
E12F066B closed
E12F064C closed
E12F021 closed

Sequence of Events:

1. Vessel drainage commences.

2. Vessel level decreases to Level 3, E12F008 and E12F009 automatically close (if interlock is operable).

3. ADHRS pumps trip on low suction pressure.

4. Vessel level decreases to Level 1, RHR 'B' and 'C' pumps receive auto initiation signals.

5. RHR 'B' pump starts (E12F004B is open). RHR 'C' pump does not start (E12F004C is closed).

6. Operator stops ADHRS, remote manually closes ADHRS flow control valve, opens E12F004C, and starts RHR 'C' pump.

Evaluation:

Initiation of Safety Functions:

- o No interference with the automatic start of the RHR 'B' pump (or manual start of the B train if not aligned for LPCI initially) or manual realignment and start of the RHR 'C' pump is indicated.
- o The two train flow paths can be established, and other paths isolated similar to the discussion provided in Case 3.1 above.
- o No suction or discharge bypasses are indicated.

Operability During Standby:

Comments in Case 3.1 apply, including non-LPCI mode alignment of RHR 'C'.

Minimum Flow: No interference with E12F064B or E12F064C function indicated.

Inadvertent Drainage:

- o See Table 1 for initial lineup.
- o Potential backflow through a failed open ADHRS check valve stopped at E12F004C.

ADHRS Pressure Integrity:

Closure of the ADHRS flow control valve and/or closure of the discharge check valve isolates the ADHRS piping from the RHR 'C' discharge piping.

Single Failure/Operator Error:

- o If the LPCI 'C' injection valve E12F042C failed in the closed position, the ADHRS isolation check valve would isolate the high pressure RHR 'C' pump discharge from the ADHRS piping. Therefore, overpressurization of the ADHRS piping would not occur.

- o Failure of the ADHRS isolation check valve would take away isolation capability between the RHR 'C' discharge piping and the ADHRS discharge piping. Backflow to the RHR suction piping would be prevented by the ADHRS pump discharge check valve. Overpressure of ADHRS would not occur since the operating pressure of the a RHR 'C' loop is lower than the design pressure of the ADHRS piping.

- o If the low level 3 isolation actuation of E12F008 and E12F009 is not operable, or if the operator actuates RHR 'C' prior to reaching level 3, parallel operation of the RHR 'C' pump and ADHRS may occur. This is because credit for securing ADHRS cannot be assumed (either due to operator error of failure of both the non-Q pump trip circuitry/breaker and the non-Q motor-operator to close the ADHRS flow control valve) and because the ADHRS pump shutoff head is greater than the RHR 'C' loop operating pressure.

  Avoidance of parallel operation can be accomplished by requiring the operator to close valves in the ADHRS suction path - E12F008, E12F009, E12F006A and E12F066A so that suction is lost and ADHRS flow ceases. More than one single failure or operator error would be necessary to defeat this protection action.

- o No operator error relative to misalignment of valves in the ADHRS or other valves in the connecting RHR trains in response to the LOCA condition is indicated that would result in adverse consequences.

  Significant Equipment Damage: None indicated.

  Conclusions: This case is considered acceptable, predicated on the procedural approach for ADHRS and RHR 'C' parallel flow protection discussed above.

3. <u>Case 3.3 and 3.4 Evaluations</u>

Initial Conditions: Same as Case 3.1 - ADHRS cooling operation via E12F006A and E12F066A.

For Case 3.3, RHR A and B trains are assumed to be the operable ECCS trains.

For Case 3.4, RHR A and C trains are assumed to be the operable ECCS trains.

These represent "abnormal" combinations in that the assumed operable ECCS trains are from different divisions, which is generally unlikely.

In Case 3.3, the evaluation of Case 3.1 is relevant to the consequences of a LOCA to RHR 'A' train evaluation, and Case 3.2 is relevant to 'B' train operation.

In Case 3.4, the evaluation of Case 3.1 is relevant to the 'A' train and Cases 3.2 and 3.6 are relevant to the 'C' train.

4. Case 3.5, 3.6, 3,7 and 3.8 Evaluations

Initial Conditions: ADHRS in vessel to vessel cooling via E12F006B/E12F066B with LOCA event.

E12F004B closed
E12F008 and E12F009 open
E12F006B open
E12F066B open
E12F064B closed
E12F064C closed
E12F021 closed

Assumed operable ECCS:

Case 3.5:	RHR A + LPCS
Case 3.6:	RHR B and C
Case 3.7:	RHR A and B
Case 3.8:	RHR A and C

The difference in these cases from 3.1 - 3.4 is that ADHRS flow is through E12F006B/E12F066B rather than E12F006A/E12F066A. the various conclusions of Cases 3.1 - 3.4 apply to these. No different effects are indicated when the 'B' pump is postulated to fail and the 'C' RHR loop is the second ECCS train rather than LPCS in Case 3.1.

It should be noted that for Case 3.6, the possibility of parallel flow of ADHRS with RHR 'C' LPCI flow exists as discussed in Case 3.2. In this case, E12F006B must be closed to allow E12F004B to be opened for the LPCI suction path which will prevent ADHRS flow from continuing. If E12F006B fails to close, a procedural requirement to close E12F066B prior to RHR use will prevent ADHRS flow from continuing.

This situation is also encountered in Case 3.8, and the procedural requirements in Case 3.2 are pertinent.

5. Case 3.9, 3.10, 3.12 and 3.13 Evaluations

Initial Conditions: ADHRS in vessel to vessel cooling with LOCA event.

Case 3.9 and 3.10: Same as Case 3.1
(flow via E12F006A and E12F066A)
Case 3.12 and 3.13: Same as Case 3.5
(flow via E12F006B and E12F066B)

Assume Operable ECCS:

Case 3.9:	RHR A
Case 3.10:	RHR B
Case 3.12:	RHR A
Case 3.13:	RHR B

These all involve ADHRS operation in Mode 5 with the cavity flooded. Per the GGNS Technical Specifications, ECCS capability is not required in this circumstance. However, GGNS policy calls for at least one "functional" ECCS train in TSPS 109. These cases reflect that policy.

The evaluations in Cases 3.1 and 3.2 apply for the applicable ADHRS suction path (E12F006A/E12F066A or E12F006B/E12F066B) and assumed operable ECCS train. The logic presented in Case 3.1 for the acceptablity of a failure or misposition of the E12F066A/B permissive bypass does not apply here as there is not necessarily a redundant functional ECCS loop. The failure of the permissive bypass switch will be mitigated by administratively requiring the E12F066A/B valve to be closed prior to manually starting the RHR A/B pump (and opening the E12F004A/B valve).

6. Case 3.11 and 3.14 Evaluations

Initial Conditions: ADHRS in vessel to vessel cooling with LOCA event.

Case 3.11: Same as Case 3.1 (flow via E12F006A and E12F066A)

Case 3.14: Same as Case 3.5 (flow via E12F006B and E12F066B)

Assumed Operable ECCS: RHR C

These involve the use of only the 'C' RHR train for ECCS in Mode 5 with the cavity flooded.

Procedural requirements for avoidance of parallel flow with ADHRS, as discussed earlier in this Table Case 3.2, apply.

TABLE 6

CASE 4 EVALUATIONS
(ADHRS in Spent Fuel Pool to Vessel Cooling with a LOCA)

1. Case 4.1 Evaluation:

Initial Conditions: ADHRS operating in spent fuel pool to vessel mode with a LOCA

G41F226 open
G41F348 open
E12F004A open or closed
E12F006A closed
G41F059, E12F066A, E12F066B closed

Assumed functional ECCS: RHR A

Sequence of Events (No Operator Action):

1. Pool drainage commences.

2. Pool drains below ADHRS suction point, ADHRS pumps trip on low suction pressure.

3. Level decreases to Level 1, RHR 'A' pump receives auto-initiation signal. Permissive for RHR 'A' pump allows pump start as E12F004A is open.

Sequence of Events (Operator Action):

1. Pool drainage commences.

2. Operator stops ADHRS pumps, shuts discharge valve.

3. Operator starts RHR 'A' pump remote manually (opens E12F004A first, if required).

Evaluation:

Initiation of Safety Functions:

- No interaction between ADHRS and the RHR 'A' loop initiation circuitry is indicated.

- No interaction between ADHRS and various parts of RHR 'A' train flow path is indicated.

- No LPCI flow bypasses are indicated. ADHRS is isolated from the RHR 'A' train by two closed isolation valves (G41F059 and E12F066A).

Operability During Standby:

- o Keep fill function of RHR 'A' maintained as discussed in Section II.D.2.a.3.

- o Alignment does not interfere with minimum flow portion of RHR 'A'.

Minimum Flow: No interference with E12F064A function indicated.

Inadvertent Drainage:

- o See Table 1 for initial lineup.

- o Due to ADHRS suction point on spent fuel pool and double isolation from RHR loops, no drainage as a result of LOCA response is indicated.

ADHRS Pressure Integrity: No interface between high pressure source (RHR 'A' discharge pressure) and ADHRS exists.

Single Failure/Operator Error: No interface with the ADHRS is involved with this lineup requiring automatic actuations or operator action. ADHRS is isolated from RHR 'A' by two closed isolation valves (G41F059 and E12F066A), neither of which has any automatic opening signal, and only one of which is remote manually operable.

As postulated in Table 5, Cases 3.1 and 3.2, the ADHRS may continue to operate due to failure of the non-Q pump control circuitry/discharge valve motor operator. In this case, if the ADHRS suction is uncovered as the pool drains, the pump may trip on low discharge pressure, but in any case will not continue to pump to the vessel after this level is reached. As this level is a "safe" level per UFSAR Section 9.1, no adverse consequences are identified.

Significant Equipment Damage: None indicated.

Conclusions: This case is acceptable.

2. Case 4.2 Evaluation:

Initial Conditions: ADHRS operating in fuel pool to vessel mode with a LOCA

G41F226 open
G41F348 open
E12F004B open or closed
E12F006B closed
G41F059, E12F066A and E12F066B closed

Assumed functional ECCS: RHR B

Due to the symmetry between the A and B loops of RHR, and with the interface with ADHRS, the results of the Case 4.1 evaluation and conclusions apply.

3. <u>Case 4.3 Evaluation</u>:

Initial Conditions: ADHRS in fuel pool to vessel mode with a LOCA

Valve alignment from spent fuel pool:

Same as Cases 4.1 and 4.2

Assumed functional ECCS: RHR C

This case is analogous to Case 3.2 in Table 5 in that the suction source of ADHRS does not affect the plant response.

As identified in Table 5, Case 3.2, failure of the ADHRS to stop on demand would result in parallel flow of the ADHRS with RHR 'C' LPCI flow (at least until the ADHRS suction is uncovered). However, unlike Case 3.2, remote manual means to isolate the ADHRS suction are not available to terminate ADHRS flow. However, in these circumstances, an ECCS capability is not required by the Technical Specifications. Satisfaction of TSPS 109 for ECCS functionality is considered to be met in this case by local action, if necessary, to terminate ADHRS flow prior to start of the C pump.

4. <u>Case 4.4 through 4.7 Evaluations</u>

Initial Conditions: ADHRS in spent fuel pool to vessel mode with a LOCA

Valve Alignments:

Case 4.4: Same as Case 4.1 except E12F006A is open, E12F004A closed

Case 4.5: Same as Case 4.2 except E12F006B is open, E12F004B closed

Case 4.6: Same as Case 4.1 except E12F004A is open

Case 4.7: Same as Case 4.2 except E12F004B is open

Assumed functional ECCS:

Case 4.4: RHR A, operating in SDC
Case 4.5: RHR B, operating in SDC
Case 4.6: RHR A, operating in SDC
Case 4.7: RHR B, operating in SDC

In these cases, the functional ECCS loop (RHR A or B) is assumed to be either in the shutdown cooling or suppression pool cooling modes prior to the event as allowed by the evaluation in Section II.D.2.g (however unlikely this may be). As seen in the scenario in Case 4.1, the system response to the LOCA event does not depend on the initial operating state of the operable ECCS for these two modes, nor do they involve any interconnection in the ADHRS fuel pool to RPV cooling mode. Therefore, these are considered acceptable.

TABLE 7

CASE 5 EVALUATIONS
(ADHRS in Flush Mode with a LOCA)

1. Case 5.1, 5.3, 5.5 and 5.6 Evaluations:

Initial Conditions: ADHRS in flush mode with a LOCA

ADHRS alignment:

E12F004C open
G41F057 and E12F066C open
E12F021 open
E12F064C closed
G41F059, E12F066A and E12F066B closed

Case 5.1: RHR A aligned for SDC (E12F006A open) E12F004A closed.

Case 5.3: RHR A or B aligned for SDC, RHR B or A in standby (in standby, E12F004 valve may be open, E12F064 valve may be open).

Case 5.5: RHR A aligned for SDC.

Case 5.6: RHR B aligned for SDC (E12F006B open, E12F004B closed).

These cases differ from Cases 4.1 and 4.2 only in that the ADHRS suction is from the suppression pool. Double valve isolation is maintained between the ADHRS and RHR loops A and B, and no functional interactions are involved. Consequently these cases are considered acceptable.

2. Case 5.2 Evaluation:

Initial Conditions: ADHRS operating in flush mode with a LOCA

E12F004C open
E12F004B closed
E12F006B open
G41F057 and E12F066C open
G41F059, E12F066A and E12F066B closed
Cavity drained
RHR 'B' in SDC, RHR 'C' in standby
E12F021 open
E12F064C closed

Sequence:

1. Vessel drainage commences.

2. Vessel level decreases to Level 3, E12F008 and E12F009 automatically close.

3. Vessel level decreases to Level 1, RHR 'B' and 'C' pumps receive auto initiation signal, RHR 'B' pump does not start because of suction valve permissives.

   RHR 'C' pump starts (E12F004C open permissive allows pump start).

4. Operator stops ADHRS pumps and shuts discharge valve.

5. Operator opens E12F004B, realigned systems for LPCI, and starts RHR 'B' pump.

Evaluation:

Initiation of Safety Functions:

- No interference is indicated due to ADHRS lineup or operation.

- No interference with the RHR 'A' or 'B' pump flow paths is indicated.

- No flow bypasses are indicated. RHR 'C' short circuiting back to its suction through ADHRS is precluded by the ADHRS discharge check valve and the check valves on each ADHRS pump discharge, allowing a single failure.

Operability During Standby:

- Keep fill function maintained as discussed in Section II.D.2.a.3.

- No interference with minimum flow function of RHR 'B'.

  Depending on ADHRS flow rate, RHR 'C' minimum flow valve E12F064C may not open when RHR 'C' pumps starts. However, since valve E12F021 will be open for ADHRS flow, minimum flow protection is not required.

Inadvertent Drainage:

- See Table 1 for initial lineup.

o No potential drainage path created by system response: No interconnection with 'B' RHR loop, ADHRS is connected to the suppression pool.

Code Class Boundary: The ADHRS flush lineup requires valves E12F057 and E12F066C to be open. In this situation, the ASME Class 2 RHR 'C' LPCI suction path is not isolated from the Class 3 piping beyond G41F057. As neither of these valves are remote manually isolable from the control room, the RHR 'C' loop would not be operable for this case, however, it would be functional. Note that operability of the RHR 'C' loop is only considered for the purposes of this study (Case 5.2). Other ECCS may be operable, and the RHR 'C' loop would not be required.

ADHRS Pressure Integrity:

ADHRS discharge check valve isolates ADHRS from RHR 'C' discharge piping.

Single Failure/Operator Error:

o Failure of the ADHRS isolation check valve would take away isolation capability between the RHR 'C' discharge piping and the ADHRS discharge piping. Backflow to the RHR suction piping would be prevented by the ADHRS pump discharge check valves. Overpressure of ADHRS would not occur since the operating pressure of the RHR 'C' loop is lower than the design pressure of the ADHRS piping.

o Failure of RHR 'C' pump to start - since E12F004C was already open, the loop reverts to the condition it was in before the event.

o Operator intervention before automatic initiations (and/or Level 3 interlock not operable) - Since the only operator action relative to ADHRS is to start the RHR 'C' pump and secure ADHRS, no unacceptable consequences are identified. (As the E12F042C is normally closed prior to starting the RHR 'C' pump the sequence of actions specified in Case 3.2 in Table 5 applies).

o The potential for continued ADHRS operation in parallel with RHR 'C' exists, as described in previous cases. Unlike previous cases, both remote manual closure of the suction to stop ADHRS flow and potential relaxations of criteria (as discussed in Case 4.3) are unavailable.

Significant Equipment Damage: None indicated.

Conclusions: This case is not acceptable based on the inability to remote manually isolate non-ASME Clase 2 portions of the RHR 'C' LPCI suction piping and to stop parallel ADHRS flow. The RHR 'C' loop may not be used for ECCS when ADHRS is in the flush mode as described above.

3. <u>Case 5.4, 5.7 and 5.8 Evaluations</u>:

Initial Conditions: ADHRS in flush mode with a LOCA

ADHRS alignment: same as shown for Cases 5.1, 5.3, 5.5, 5.6

Case 5.4: RHR A aligned for SDC
Case 5.7: RHR A or B aligned for SDC
Case 5.8: RHR A or B aligned for SDC

These all involve the RHR 'C' train, and the evaluation provided for Case 5.2 above is valid in these instances, with the exception of the ADHRS overpressurization protection in Case 5.7 and 5.8, for which the discussion Cases 3.11/3.14 in Table 5 applies, then.

TABLE 8

CASE 6 EVALUATIONS
(ADHRS in Vessel to Vessel Cooling · th Loss of SDC)

1. Case 6.1 Evaluation:

Initial Conditions: RHR 'A' operating in SDC mode with
ADHRS in standby Mode 4
E12F006A open
E12F008 and E12F009 open
E12F066A and E12F066B closed
G41F059 open
E12F004C closed
E12F424 closed
E12F064C closed
Vessel head on
RHR 'B' in maintenance

Sequence of Events:

1. RHR 'A' SDC mode fails (e.g., pump motor fails).
2. Vessel temperature (and vapor pressure) increases.
3. Operator opens E12F066A remote manually. Opens ADHRS discharge valve, starts ADHRS pumps and opens RHR vessel injection valve E12F042C.

Evaluation:

Initiation of ADHRS Functions: No interferences or bypasses are indicated.

Operability During Standby: As relates to ADHRS, there is no interaction from the RHR 'A' loop that would prevent the actuation of the ADHRS.

The ADHRS and RHR 'A' are separated at E12F066A.

Inadvertent Drainage: See Table 1 for initial lineup. Opening of valve E12F066A does not introduce a new drain path.

ADHRS Pressure Integrity: No effects indicated.

Single Failure/Operator Error:

- o Various single failures may defeat the initiation of ADHRS, primarily motor-operated valve failures to open. See discussion below.

- o If the operator fails to open E12F066A prior to starting the ADHRS pumps, the pumps will run with no suction path unless the low suction pressure trips the pumps (this is not a safety function to trip the pumps).

- o If the operator fails to open the ADHRS discharge valve, the ADHRS pumps will run with no discharge path.

Significant Equipment Damage:

Damage to the ADHRS pumps due to a lack of a suction path would be prevented by the low suction pressure trip. However, damage to the pumps could occur if the trip function failed. Also, pump damage could occur if a discharge path was unavailable. Such postulated damage would be limited to ADHRS components.

Discussion:

As noted above, various single failures could be postulated to prevent failure of motor-operated valves such as E12F066A or E12F042C to close or the ADHRS discharge valve to open, the ADHRS pumps to start, or PSW failure.

Relative to safety-related valves, these represent no greater probability of failure than would valves in RHR train 'B' if it were in standby and actuated. Regarding active components of the ADHRS, which are not safety-related, it is considered that they do not represent a greater probability of failure either.
Any of the motor-operated valves are capable of local manual operation with a handwheel if the failure is in the motor-operator, its power supply, or control circuitry. Depending on the reactor decay heat rate, and PSW temperature and flow, the ADHRS pumps may be redundant, such that the failure of one pump to start may be inconsequential.

Conclusions: Based on the assumption that ADHRS is only used in Mode 4 or Mode 5, this case is considered acceptable.

2. Case 6.2 Evaluation:

This is similar to Case 6.1, except that RHR 'B' is the operating train, and ADHRS initiation is via E12F006B/E12F066B. The evaluations, discussions, and conclusions of 6.1 apply.

3. Case 6.3 and 6.4 Evaluations:

Initial Conditions: RHR 'A' or 'B' operating in SDC mode with ADHRS in standby in Mode 5.

System/Valve Alignment:

Case 6.3: Same as Case 6.1
Case 6.4: Same as Case 6.2

These are similar to Cases 6.1 and 6.2 except that the vessel is in mode 5 (with cavity either drained or flooded). These cases are considered acceptable.

4. Case 6.5 Evaluation:

Initial Conditions: ADHRS operating in vessel to vessel mode via E12F006A/E12F066A with RHR 'A' SDC train in standby, in Mode 5

E12F008 and E12F009 open
E12F006A open
E12F004A closed
E12F064A closed
E12F066A open
G41F059 open
E12F004C closed
E12F424 open
E12F021 closed
E12F064C closed

Sequence of Events:

1. ADHRS cooling function lost.

2. Operator shuts E12F066A and secures ADHRS.

3. Operator starts RHR 'A' pump and establishes flow in the RHR 'A' loop.

Evaluation:

Initiation of Safety Function: No inhibitions of RHR 'A' pump start or valve lineups are indicated. No RHR SDC bypasses are indicated.

Operability During Standby/Minimum Flow:

- o Keep fill capability acceptable per II.D.2.a.3.

- o Alignment does not impact RHR 'A' minimum flow capability during standby or initiation.

Inadvertent Drainage:

- o See Table 1 for initial lineup.

- o Realignment places RHR 'A' in current plant design condition relative to RHR 'A' loop.

ADHRS Pressure Integrity: No interface between high pressure source (RHR 'A' discharge pressure) and ADHRS exists.

Single Failure/Operator Error:

If the operator fails to close E12F066A or E12F066A fails to close, the RHR 'A' SDC suction is not isolated from the ADHRS suction path. No functional problems will ensue since backflow from the ADHRS to the RHR 'A' suction is prevented by 3 check valves in series (E12F041C, E12F416, E12F412A and B.

To the extent that operability of RHR 'A' SDC is compromised by connection to an ASME Section III Class 3 system, isolation can be accomplished by local manual closure of valve E12F066A or E12F059. Such local manual action is acceptable per Section II.D.2.a. No specific time constraints are indicated for such action, although a temperature excursion out of mode 5 (140 °F) may be possible, however, this is just as likely to occur due to a problem in initiating the RHR backup train when an operating RHR SDC train was in operation as when the ADHRS was in operation; no greater probabilities or consequences are indicated.

Significant Equipment Damage: None indicated.

Conclusions: This case is acceptable.

5. <u>Case 6.6 Evaluation</u>:

Initial Conditions: ADHRS operating in vessel to vessel mode via E12F006B/E12F066B with RHR 'B' train in standby, in Mode 5.

System alignment, same as Case 6.5 except E12F006A/E12F066A closed, E12F006B/E12F066B open, E12F004B and E12F064B closed.

Case 6.6 is identical to Case 6.5, except that ADHRS flow is through E12F006B/E12F066B, and the backup RHR SDC train is loop 'B' The evaluations and conclusions of Case 6.5 above apply.

6. <u>Case 6.7 and 6.8 Evaluations</u>:

Initial Conditions:

Case 6.7: ADHRS in vessel to vessel mode via E12F006A/E12F066A, with RHR 'B' train in standby in Mode 5.

Case 6.8: ADHRS in vessel to vessel mode via E12F006B/E12F066B, with RHR 'A' train in standby in Mode 5.

System Alignments:

Case 6.7: Same as Case 6.5
Case 6.8: Same as Case 6.6

Case 6.7 and 6.8 indicate ADHRS flow through the RHR SDC train valves opposite the backup loop.

The evaluations and conclusions of Case 6.5 apply to these cases.

7. Case 6.9 Evaluation:

Initial Conditions: ADHRS operating in vessel to vessel mode via E12F006A/E12F066A with RHR 'A' train in standby, in Mode 4

E12F008 and E12F009 open
E12F006A open
E12F004A closed
E12F0064A closed
E12F066A open
G41F059 open
E12F004C closed
E12F424 open
E12F021 closed
E12F064C closed

Sequence of Events:

1. ADHRS cooling function lost.

2. Operator shuts E12F066A and secures ADHRS.

3. Operator starts RHR 'A' pump and establishes flow in the RHR 'A' loop.

Evaluation:

Initiation of Safety Function: No inhibitions of RHR 'A' pump start or valve lineups are indicated. No RHR SDC bypasses are indicated.

Operability During Standby/Minimum Flow:

- o Keep fill capability acceptable per II.D.2.a.3.
- o Alignment does not impact RHR 'A' minimum flow capability during standby or initiation.

Inadvertent Drainage:

- o See Table 1 for initial lineup.
- o Realignment places RHR 'A' in current plant design condition relative to the RHR 'A' SDC loop.

ADHRS Pressure Integrity: No interface between high pressure source (RHR 'A' discharge pressure) and ADHRS exists.

Single Failure/Operator Error:

If the operator fails to close E12F066A or E12F066A fails to close, the RHR 'A' SDC suction is not isolated from the ADHRS suction path. No functional problems will ensure since back flow from the ADHRS to the RHR 'A' suction is prevented by 3 check valves in series (E12F041C, E12F416, and E12F412A or B).

To the extent that operability of RHR 'A' SDC is compromised by connection to an ASME Section III, Class 3 system, isolation can be accomplished by local manual closure of valve E12F066A or E12F059. Failure to isolate the ADHRS suction piping could result in overpressurization if RHR 'A' SDC operation is not initiated. However, based on decay heat loads it is judged that adequate time exists for operator action to isolate the ADHRS suction piping before design pressures are exceeded. In addition, procedural requirements to isolate ADHRS suction piping whenever reactor temperature exceeds 200 degrees or ADHRS is not in operation will be implemented to ensure that design pressures are not exceeded.

Significant Equipment Damage: None

Conclusions: This case is acceptable.

8. Case 6.10 Evaluation:

Initial Conditions: ADHRS operating in vessel to vessel mode via E12F006B/E12F066B with RHR 'B' train in standby, in Mode 4.

E12F008 and E12F009 open
E12F006B open
E12F004B closed
E12F064B closed
E12F066B open
G41F059 open
E12F004C closed
E12F424 open
E12F021 closed
E12F064C closed

Case 6.10 is identical to Case 6.9 except that ADHRS flow is through E12F006B/E12F066B, and the backup RHR SDC train is loop 'B'. The evaluations and conclusions of Case 6.9 above apply.

TABLE 9

CASE 7 EVALUATIONS
(ADHRS in Spent Fuel Pool to Vessel Cooling with Loss of SDC)

1. Case 7.1 Evaluation:

Initial Conditions: ADHRS in spent fuel pool to RPV cooling mode, RHR 'A' in backup SDC mode in standby

G41F226	open
G41F348	open
G41F059	closed
E12F066A	closed
E12F066B	closed
E12F004C	closed
E12F064C	closed
E12F021	closed

Sequence of Events:

1. ADHRS cooling function lost.

2. Operator opens G41F059 (locally) and E12F066A (remote manually) starts RHR 'A' pump and establishes RHR SDC flow path to fuel pool.

3. Operator secures ADHRS (portion that did not fail, e.g., closes discharge valve).

Evaluation:

Initiation of Safety Functions:

- o No interference with RHR 'A' SDC initiation or flow path is indicated.

- o Potential bypass of suction source of RHR 'A' from ADHRS is precluded by three check valves in series (E12F041C, E12F416, and E12F412A or B).

Operability During Standby:

- o Keep fill function maintained as discussed in II.D.a.3.

- o No other effects of ADHRS operation are indicated for RHR loop A during standby.

Minimum Flow: No interference with E12F064A function indicated.

Inadvertent Drainage:

- o See Table 1 for initial lineup.

- o As the ADHRS pump and RHR 'A' pump both take suction on the spent fuel pool, and are not connected to the RPV, inadvertent drainage from RPV paths is not indicated. To the extent that valve E12F064A will open (if initially closed) to provide pump minimum flow when the RHR 'A' pump starts and the discharge path is opening, some drainage may occur to the suppression pool. However, such a condition is not related to the prior ADHRS operation or alignment and is not evaluated further.

ADHRS Pressure Integrity: No interface between a high pressure source (RHR 'A' discharge pressure) and ADHRS exists.

Single Failure/Operator Error:

- o Failure of E12F066A or G41F059 to open - this may defeat the ability of the RHR 'A' loop to assume cooling duty from the spent fuel pool, assuming local manual operation of E12F066A and G41F059 is not successful. However, this is no more probable than existing design cases in which RHR 'A' may be a backup/standby to the RHR 'B' train. In any event, with the cavity/upper containment pool flooded, a backup capability to an operable RHR SDC train is not required.

- o Operator error to properly align the RHR 'A' train for SDC. As this is not a function of the ADHRS alignment, it is not specifically considered, and in any event, would be recoverable by corrective action to make the proper alignment.

Conclusions: This case is acceptable.

2. <u>Case 7.2 Evaluation</u>:

Initial Conditions: ADHRS in spent fuel pool to RPV cooling mode, RHR 'B' in backup SDC mode in standby.

System Alignment: Same as Case 7.1

This case assumes the RHR 'B' train is available as a backup. Due to the symmetry between the RHR 'A' and 'B' loops, and their interconnection with ADHRS, the results of Case 7.1 above are applicable.

ATTACHMENT 2

Summary of References Cited in Safety Evaluation
for ADHRS Modification (DCP 88/0008)

1. GGNS UFSAR, Revision 2

Section 3.2
Section 3.3
Section 3.4
Section 3.5
Section 3.5.1.3
Section 3.5.1.4
Table 3.5-1
Section 3.6
Section 3.6.2
Section 3.6A.1.1
Section 3C.3.2
Section 3.C.4.2.6
Section 3.9
Section 3.11.4
Section 3.11.5.3
Table 3.11-1
Table 3.11-2
Section 3.9
Section 5.4.7.2.7
Section 6.2
Chapter 8
Section 9.1
Section 15.1.4
Section 15.1.6
Section 15.2.6
Section 15.2.10
Section 15.7.2
Section 15.7.3
Table 15.7-7

2. GGNS Unit 1 Technical Specifications

Table 3.3
Table 3.3.2-1
Table 3.3.7.5-1
TS 3/4.4.9.1
TS 3/4.4.9.2
TS 3/4.4.11
TS 3/4.5.2
TS 3/4.5.3
TS 3/4.7.9
TS Bases 3/4.4.9
TS Bases 3/4.4.11
TS Position Statement 109

ATTACHMENT 2

Summary of References Cited in Safety Evaluation
for ADHRS Modification (DCP 88/0008)
(Continued)

3. GGNS Fire Hazards Analysis Report

4. Others

GE Process Diagram 762E425BA, Rev. 8
SFD-1085, Rev. 3
10CFR50 Appendix R
Crane Technical Paper 410 (1957)
NUREG 0800, Rev. 1, SRP 15.6.5 Appendix B
Regulatory Guide 1.26, Rev. 3
Regulatory Guide 1.29, Rev. 3
Various P&IDs and electrical schematics indicated in
Tables 1 and 2