NEW YORK POWER AUTHORITY
JAMES A. FITZPATRICK NUCLEAR POWER PLANT

EVALUATION OF THE DECAY HEAT REMOVAL SYSTEM

Report no. JAF-RPT- DHR-02413

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EXECUTIVE SUMMARY

This study report is prepared in accordance with Design Control Manual 7, "Preparation of Technical Studies and Reports" to document design bases, supporting analyses, safety considerations and operational limitations associated with installation of a Decay Heat Removal (DHR) system at the J. A. FitzPatrick Nuclear Power Plant.

Revision no. 4 of this report is being developed in accordance with Design Control Manual 7A, "Preparation of Technical Studies and Reports" (JAF only).

The results of this study are:

(1) Installation and operation of the DHR will constitute a significant enhancement in decay heat removal capability, particularly during refueling outages.

(2) Installation and operation of the DHR will improve the ability to control refueling cavity and spent fuel pool water temperature during refueling operations.

(3) Installation and operation of the system will eliminate current restrictions on fuel movement which are tied to existing spent fuel pool decay heat removal capacity.

(4) There are no unreviewed safety questions associated with the DHR and revision to the plant Technical Specifications are not required to reflect system installation and operation. Therefore, system design, installation, and operation can be evaluated pursuant to 10 CFR 50.59.

(5) Installation and operation of the DHR will provide greater flexibility in outage planning and has the potential to reduce refueling outage length.

(6) Under specified conditions, and with the appropriate administrative controls in place, both trains of RHR can safely be removed from service and the DHR utilized to remove decay heat from both the spent fuel pool and the reactor pressure vessel.

Revision no. 4 to this report has been developed to document resolution of Plant Operating Review Committee (PORC) and NRC questions raised during initial system testing and operation during refueling outage (RO) 12, to summarize system operating experience during RO 12 , and to confirm DHR design and safety criteria will be met if and when the spent fuel pool storage capacity is increased from the current limit of 2797 assemblies to 3247 assemblies. In addition, the results of a revised minimum temperature analyses for the spent fuel pool are incorporated via revision no. 4. The technical changes associated with revision no. 4 to this report are also being incorporated, as appropriate, into revision no. 4 to Nuclear Safety Evaluation JAF-SE-96-042. In order to maintain the proper grammatical tense within the body of the report, the information provided under revision no. 4 appears as additions to the text, as opposed to revisions of existing text. Hence it is necessary to read the entire text of a report section in order to understand the current status of the analyses which support DHR operation.

TABLE OF CONTENTS

Section	Subject
1	**Introduction**
2	**Design Bases Considerations**
	2.1 Heat Removal Capability
	2.2 Interactions with existing Plant Systems
3	**Description of the Decay Heat Removal System**
	3.1 Mechanical Features
	3.1.1 Normal Operating configuration
	3.1.2 Other Operating Configurations
	3.1.3 Primary - to - Secondary Leakage Considerations
	3.1.4 Location of Major Components
	3.1.5 System "Layup" Considerations
	3.2 Instrumentation and Control Features
	3.3 Electrical Features
	3.4 Civil/Structural Features
4	**Engineering and Design Issues**
	4.1 Single Failure Considerations and Overall Heat Removal Capability
	4.1.1 Phase 1: From Plant Shutdown to Removal of the SFP Gates
	4.1.2 Phase 2: SFP Gates Removed and the Total Decay Heat Load Exceeds DHR Capability in the Nominal Configuration
	4.1.3 Phase 3: SFP Gates Removed and the Total Decay Heat Load is Less Than DHR Capability in the Nominal Configuration
	4.2 Seismic Design
	4.3 System Power Supply
	4.4 Electrical Separation
	4.5 Operational Radiation Fields
	4.6 Fire Protection Issues
	4.7 Wind Loadings on Exterior Components
	4.8 Internal Flooding
	4.9 Heavy Loads
	4.10 Decay Heat Load Calculations

5 **Nuclear Safety Considerations**

5.1 Maintenance Of Spent Fuel Pool Water Inventory
5.2 Natural Circulation Cooling of Fuel Assemblies in the Reactor Pressure Vessel
5.3 Potential for Excessive Cooling of the Spent Fuel Pool
5.4 Technical Specification and Design Bases Document Review
5.5 UFSAR Review

6 **System Installation and Modification Acceptance Testing**

7 **Initial Thermal Performance Testing and Verification of Natural Circulation**

7.1 Thermal Performance Testing During RO12
7.2 Verification of Natural Circulation Cooling During RO12
7.3 System Operating Experience During RO12

8 **System Operation and Testing**

8.1 DHR Operation with the Plant on line
8.2 DHR Operation During Refueling Outages - SFP Gates Installed
8.3 DHR Operation During Refueling Outages - SFP Gates Removed
8.4 Pre-outage System Testing

9 **Additional Supporting Analyses**

9.1 Evaluation of Single Active failures of DHR Components, and Prospective Mitigating Actions with the System Operating in the Nominal Heat Removal Configuration
9.1.1 Postulated Active Failure of a Mechanical Component
9.1.2 Postulated Active Failure of an Electrical, Instrument, or Control Component
9.1.3 Evaluation of Prospective Mitigating Actions
9.1.4 Time Frame for the Implementation of Mitigating Actions
9.2 Estimated time to Boiling Given A Loss of all Decay Heat Removal
9.3 Radiological Consequences of Boiling
9.4 DHR System Reliability Assessment
9.5 Effects of Elevated Water Temperature on SFP Components

10 **Conclusions and Recommendations**

List of Tables

1 Decay Heat Loads
2 Decay Heat Removal Capabilities
3 Dose Consequences Following a Postulated Loss of Decay Heat Removal

List of Figures

1 DHR System Flow Diagram - Sheet 1 of 2
2 DHR System Flow Diagram - Sheet 2 of 2
3 DHR System Electrical One-line Diagram
4 Decay Heat Versus Time After Shutdown; Core and Spent Fuel Pool
5 DHR Performance During RO12

List of References

1 Design Control Manual 7, "Preparation of Technical Studies and Reports", Revision 0

2 Code of Federal Regulations, Title 10, Part 50, subpart 59, "Changes, tests and experiments"

3 IE Bulletin 80-10, "Contamination of Non-radioactive Systems and Resulting Potential for Unmonitored and Uncontrolled Release to the Environment", USNRC

4 NYPA Calculation no. JAF-CALC-MISC-02244, "Assessment of the Combined Decay Heat Load of the Reactor Core and Spent Fuel Pool during Refueling Outages", Revision 1

5 NRC Letter, B. C. McCabe to R. E. Beedle (NYPA), Issuance of Amendment for James A. FitzPatrick Nuclear Power Plant (TAC No. M76937)", dated 31 december 1991 (Note: This is Technical Specification Amendment no. 175 and contains the NRC SER for the most recent spent fuel pool storage expansion).

6 Safety Evaluation for the James A. FitzPatrick Nuclear Power Plant, Docket No. 50-333, U. S. Atomic Energy Commission, Directorate of Licensing, Issued 20 november 1972

7 JAF Nuclear Safety Evaluation no. JAF-SE-90-042, "General Heavy Load Handling System Requirements to Meet NUREG-0612 Criteria", Revision 0.

8 JAF Maintenance Procedure MP-088.01, "Heavy Load Handling", Revision 12.

9 NUREG-0800, USNRC Standard Review Plan, Section 9.2.5, "Ultimate Heat Sink", Revision 2, dated july 1981 (Note: Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling", is appended to this section of the SRP).

10 Significant Operating Experience Report 85-1, "Reactor Cavity Seal Failure", INPO, January 1985 (and NYPA's evaluation thereof).

11 General Electric Nuclear Energy Letter, F. Paradiso/H. Choe to K. A. Phy (NYPA), "Decay Heat Removal System Project, Natural Circulation Heat Transport Calculation Results", dated 18 april 1996

12 NYPA Calculation no. JAF-CALC-DHR-02380, "Alternate Decay Heat Removal System Thermal-Hydraulic Analysis", Revision 2

13 General Electric Nuclear Energy Letter W. H. Brown to D. Lindsey (NYPA), "ADHR Questions", dated November 15, 1995

14 NYPA Design Basis Document for the Residual Heat Removal System

15 NUREG 1433, "Standard Technical Specifications for General Electric Plants, BWR/4", USNRC, April 1995

16 NYPA Calculation no. JAF-CALC-MISC-02373, "Decay Heat Management Calculation With New SFP Volume", Revision 2

17 NYPA Calculation No. JAF-CALC-RAD-00053, "Radiological Impact of Postulated Failure of the Alternate Decay Heat Removal System and Loss of Spent Fuel Cooling", Revision 0

18 Technical Information Document (TID) 14844, "Calculation of Distance Factors for Power and Test Reactor Sites", US Atomic Energy Commission, dated 23 march 1962

19 NYPA Memorandum, JAG-93-285, J. A. Gray, Jr. to H. Salmon, "Licensing Basis for Spent Fuel Pool Cooling", dated 19 november 1993 (plus attachment 1, "Loss of Spent Fuel Pool Cooling During design Basis Accident", and attachment 2, "Licensing Basis for Expansion of Spent Fuel Pool Storage Capacity, Analyzed Accidents and Assumptions for Spent Fuel Pool Cooling System")

20 NYPA Report No. JAF-ANAL-MISC-02372, "Alternate Decay Heat Removal System", Revision 0

21 NUREG-0800, USNRC Standard Review Plan, Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System", Revision 1, dated july 1981

22 JAF Administrative Procedure AP-10.09, "Outage Risk Assessment", Revision 3

23 IE Information Notice 96-39, "Estimates of Decay Heat using ANS 5.1 Decay Heat Standard May Vary Significantly", USNRC, dated 05 july 1996

24 NYPA OE/VI Evaluation of IEIN 96-39, DER 96-0888

25 JAF Nuclear Safety Evaluation JAF-SE-96-011, "Moving Fuel While MSIVs Are Being Worked With Main Steam Line Plugs Installed", Revision 1

26 JAF Abnormal Operating Procedure AOP-53, "Loss of Spent Fuel Pool, Reactor Head Cavity Well, or Dryer Separator Storage Pit Water Level*", Revision 4

27 JAF Nuclear Safety Evaluation SE-96-039, "Installation and Acceptance Testing of the Decay Heat Removal System"

28 JAF Nuclear Safety Evaluation SE-96-042, "Use of the Decay Heat Removal System in Various Plant Modes and Configurations", revision 3

29 JAF Procedure SP-01.11, "Unmonitored Paths Sampling and Analysis*", Revision 5

30 HOLTEC International Report, "Licensing Report for Reracking of J. A. FitzPatrick Spent Fuel Pool", Project no. 60864

31 NYPA Memorandum JPRJ-KV-96-174, "PORC Follow Item 96-107-01", Vehstedt to Maurer, dated 30 october 1996

32 NYPA Memorandum JPRJ-KV-96-181,"PORC Follow Item 96-107-01", Vehstedt to Maurer, dated 05 november 1996

33 NRC Inspection Report No. 50-33/96-07, transmitted via letter C. J. Cowgill to M. J. Colomb dated 13 december 1996

34 NYPA Letter JAFP-97-0018, "Reply to Notice of Violation; NRC Inspection report 50-333/96-07", dated 17 january 1997

1. Introduction

Modification F1-95-121 will install a Decay Heat Removal (DHR) System in the facility. The DHR will take suction from and discharge to the spent fuel pool (SFP). As detailed in the ensuing sections of this study, the system is designed as a non-safety system, is physically independent of existing plant equipment to the maximum extent, and is primarily intended to enhance existing decay heat removal capabilities during refueling outages with the ultimate goal of enhancing outage performance. Physical installation of the modification is expected to take place prior to refueling outage 12 (RO12) during which time the plant is expected to be in operation. Modification acceptance testing is also expected to be completed prior to RO12.

The DHR conceptual design was purchased from the Southern Company and the JAF DHR installation is functionally similar to the design of the DHR system installed at Plant Hatch. System performance information from Plant Hatch has been reviewed by NYPA personnel and is being utilized in the development of DHR operating procedures for JAF.

This study is being prepared in accordance with the requirements of DCM 7 (Reference 1) for the purpose of documenting key system design features, consolidating supporting technical information (in advance of issuance of the modification package), and evaluating whether this modification can be performed under the provisions of 10 CFR 50.59 (Reference 2).

Revision no. 4 to this report summarizes system operating experience during RO12, confirms the benefits of DHR in light of proposed expansion in SFP storage capacity, and clarifies the text based on discussions between NYPA and NRC representatives during initial system testing and operation. Revision no. 4 also incorporates the results of a revised analysis of the minimum (allowable) SFP temperature. Refer to reference nos. 33 and 34 for the NRC inspection "open" or "follow" items and the Authority's responses thereto.

2. Design Basis Considerations

The three overriding design bases considerations for the DHR are;

(1) the decay heat removal capability of the DHR must equal or exceed the combined decay heat load of irradiated fuel in the SFP and the Reactor Pressure Vessel (RPV) approximately 4.5 days post-shutdown while remaining tolerant of a wide spectrum of postulated component failures,

(2) to the extent feasible, the DHR shall be mechanically and electrically independent of existing plant systems, and

(3) the system must not adversely affect any safe shutdown function of existing plant systems.

2.1 Heat Removal Capability

The DHR system is shown schematically in figure nos 1 and 2. (Note: The figures are not design drawings and are included for information only). The nominal configuration of the system will involve operation of one primary side pump, one heat exchanger, one secondary side pump, and one set of cooling towers. The system is designed such that the system heat removal capacity in that configuration will be approximately 30 X 10^6 BTU/HR given an outside ambient wet bulb temperature of 73 ^{0}F and a primary loop heat exchanger inlet water temperature of 125 ^{0}F. The combined RPV and SFP decay heat loads have been conservatively calculated as a function of time post-shutdown and 30 X 10^6 BTU/HR corresponds to approximately 4.5 days based on best estimate decay heat loads. In support of revision no. 4 to this report, new decay heat curves were generated which included additional conservatisms so as to bound potential "worst case"(maximum heat load) conditions. The new calculations, contained in revision no. 1 to reference no. 4, indicate the 30 X 10^6 BTU/HR capability of the DHR would correspond to approximately 5.8 days post-shutdown

2.2 Interactions with Existing Plant Systems

The DHR is designed to have minimum interface with existing plant systems; in particular systems important to plant safety. With the exception of penetrations through the Reactor Building pressure boundary, connections to the condensate transfer and demineralized water systems (for primary and secondary loop fill and makeup, respectively), use of existing cable trays, "commoning" of DHR annunciator inputs into the existing Fuel Pool Cooling & Cleanup (FPCC) control panel, and the obvious interface with the SFP, the DHR is mechanically and electrically separated from existing systems. Details of the design follow. The DHR system has been designated as (new) system number 32 and the component names used in the following text reflect that designation.

3. Description of the Decay Heat Removal System

3.1 Mechanical Features

3.1.1 Normal Operating Configuration

As shown in figure nos. 1 and 2, the DHR system is comprised of a primary loop which pumps the water from the SFP through heat exchanger(s) and returns it to the SFP, and a secondary loop which removes the heat from the primary loop heat exchangers via cooling towers. The primary loop includes two 100% pumps, two heat exchangers, and appropriate isolation valves and monitoring instrumentation. There are no known detrimental effects of simultaneous operation of both primary pumps, but such operation is not anticipated as that configuration is not required in order to achieve system decay heat removal objectives.

The normal or nominal operating configuration of the system will be to have one primary pump in service, 32P-1A or 32P-1B, one of the two plate and frame heat exchangers 32E-3A or 32E-3B in service, and one of the two secondary pumps, 32P-2A or 32P-2B in service discharging to one set of cooling towers. Cooling tower fan operation would be dictated by the heat load on the system. A single strainer, 32STR-1 with an automatic backwash feature, is located in the primary loop pump discharge common header upstream of the heat exchangers to filter any solid particles in the SFP water. The strainer is provided for ALARA purposes to minimize contamination of the heat exchangers and is not required for system operation.

At system design conditions, operation of the DHR in the nominal heat removal configuration corresponds to a heat rejection rate of approximately 30 X 10^6 BTU/HR. At that heat load, each of the fans in the operating cooling towers would be running in high speed. The system automatically responds to lower (than 30 X 10^6 BTU/HR) heat loads through the fan control circuitry. With neither fan in the two operating cooling towers in service (with one primary and one secondary pump in service and an ambient wet bulb temperature of 73 ^{0}F) the system heat removal rate would be approximately 2 X 10^6 BTU/HR. Thus, range of heat removal rate associated with operation in the nominal configuration is from approximately 2×10^6 to approximately 30×10^6 BTU/HR at an ambient wet bulb temperature of 73 ^{0}F.

DHR mechanical components, including piping, are stainless steel with the exception of the cooling tower cells which are constructed of galvanized sheet metal. The materials of construction were selected to provide superior corrosion resistance, eliminate the need for secondary loop water treatment, and minimize system layup considerations.

3.1.2 Other Operating Configurations

It will be possible to operate the system with two secondary pumps in service and both heat exchangers in service. In that maximum heat removal configuration, the system is designed to remove 45×10^6 BTU/HR at a wet bulb temperature of 73 ^{0}F. Intermediate configurations are also permitted, to allow system heat removal to most closely match system heat load.

3.1.3 Primary - to Secondary Leakage Considerations

The design specifically utilizes plate and frame heat exchangers to eliminate the potential for primary to secondary leakage. Based on the design characteristics of the plate and frame heat exchangers, leakage from the primary loop into the secondary loop is not considered to be a credible event. Therefore, there is no need for installation of a radiation monitor in the DHR secondary loop. In addition, the secondary side of the DHR system will be maintained at a higher operating pressure than the primary side operating pressure (approximately 15 psid) through use of an automatic pressure control valve, 32PCV-100, in the common secondary side header on the outlet side of the heat exchangers. The design includes controls which will automatically trip an operating primary side pump if the secondary to primary pressure differential were to decrease below 10 psid.

DHR sampling will be performed in accordance with existing procedure SP-01.11, "Unmonitored Paths Sampling and Analysis*" (Reference 29). Said procedure specifically addresses the sampling considerations set forth in NRC IE Bulletin 80-10, Reference 3.

The combination of the DHR design details and JAF administrative controls are consistent with the guidance contained in NRC IE Bulletin 80-10.

3.1.4 Location of Major Components

The primary side components (pumps, strainer, heat exchangers and a control panel) are skid-mounted and will be located on Elevation 326'-9" inside the Reactor Building. The secondary side components (pumps and cooling towers) will be located on the roof of the Railroad and Truck Bay and Standby Gas Treatment Building at El. 293'-0. Exterior portions of the secondary loop piping, and the demineralized water makeup line to the cooling tower basins, will be insulated and heat-traced to provide freeze protection. In addition, immersion heaters will be provided to prevent freezing of the cooling tower basin inventory.

3.1.5 System "Layup" Considerations

It is expected the DHR system will only be aligned for service immediately prior to, during, and perhaps immediately after scheduled refueling outages, although there is a potential the system could be used during non-outage conditions to supplement or substitute for FPCC. Therefore, the design incorporates the ability to physically remove the suction and discharge pipe spools in the vicinity of the SFP and for their storage at Elevation 369-6" inside the Reactor Building. In addition, the design provides for ability to isolate and drain the exterior secondary loop components. When the system is not in service, secondary containment pressure boundary will be maintained by the (intact) pressure boundary of the portion of the DHR secondary loop inside the RB. If the DHR secondary loop pressure boundary inside the RB is not intact, e.g., for test, maintenance, or inspection activities, then the manual isolation valves in the supply and return lines to the cooling towers must be closed, or blind flanges installed.

DHR mechanical components, including piping, are stainless steel, with the exception of the cooling tower cells which are constructed of galvanized sheet metal. The system materials of construction were deliberately selected to minimize system layup and corrosion considerations.

3.2 Instrumentation and Control Features

A DHR control panel will be installed in the Reactor Building on elevation 326' 9" in the immediate vicinity of the primary side pumps and the heat exchangers. This panel will contain equipment status lights, indicators, and alarms sufficient so an operator can monitor and control system operation. Local pressure and temperature indicators will be provided in the primary and secondary loops at locations appropriate for monitoring system operation. Sensed differential across the strainer will also be used to initiate the automatic backwash feature.

Differential pressure between the primary and secondary sides of the DHR heat exchangers will also be monitored and indicated at the control panel. Sensed primary to secondary differential will also be used to position 32PCV-100 and, if the sensed differential decreases below 10 psid, trip an operating primary pump(s). A common trouble alarm will be provided for the DHR system at the FPC local panel on Elevation 326'-9" which will also alarm the FPC annunciator in the Control Room. The common trouble alarm will be actuated on any of the following conditions:

1) low flow in the primary system

2) low flow in the secondary system

3) primary loop pump trip

4) cooling tower fan high vibration

5) cooling tower basin low water level

6) transformer winding temperature - high

3.3 Electrical Features

The DHR system will be powered from a reliable, offsite power source (existing 13.2 KV switchgear J02) which will feed into a (new) 1500 KVA transformer and a (new) 480V motor control center (MCC). The 13.2 KV source is independent of the existing safety related and non-safety related power supplies to the power block. Figure 3, attached, provides the DHR system electrical one-line diagram. (Note: Figure 3 is not a design drawing and is provided for information only).

During refueling outages DHR reliability will be enhanced through the use of a portable diesel generator which will be directly connected to the DHR 480V MCC. The portable diesel generator will be sized to start and carry DHR loads for operation in the nominal configuration and would be used in the event of loss of the normal 13.2 KV supply during DHR operation. Transfer from the 13.2 KV source to the diesel generator would be accomplished manually.

Initial modification acceptance testing of the DHR will include testing of both the normal system power supply and the portable diesel generator. Such testing will include actual starting and powering of DHR loads (to support operation in the nominal configuration) from both power sources. Since the portable diesel generator to be provided during refueling outages will be a rental machine and in all likelihood a different machine will be used outage-to-outage, it will not be feasible to develop a detailed test procedure for the diesel in advance of each outage. However, the minimum testing to be performed each outage will verify the ability of each rental machine to handle DHR starting and running loads up to and including the system electrical load when operating in the nominal heat removal configuration. Prior to each refueling outage, the rental diesel will be load tested by the supplier, and witnessed by NYPA representative(s), to 750 A for a minimum of one hour.

3.4 Civil/Structural Features

The secondary side DHR piping will penetrate the south wall of the Reactor Building at approximate elevation 310' 5" and connect to the equipment to be located on the roof of the Railroad and Truck Bay and Standby Gas Treatment Building. The design of the penetrations, including classification of associated piping and supports, will be consistent with existing design standards for Reactor Building penetrations. (Refer to UFSAR Table nos. 5.3-1 and 7.2-2 for tabulations of existing RB penetrations and associated environmental parameters for each penetration, respectively). The penetrations will be classified as QA Category I including the face plates, face plate anchorages to the Reactor Building wall, and the welds between the face plates and the process piping consistent with the classification of the Reactor Building itself. Also, a missile shield is being provided to protect the penetrations from tornado-generated missiles.

DHR piping and equipment within the Reactor Building will be supported to meet the QA Category II/III, Seismic Class II design criteria; that is, a failure of a DHR component or pipe during a postulated design basis earthquake will not prevent a safety-related component or pipe in the vicinity from performing its required safety-related function. The remainder of the piping, components, equipment, and supports is classified as QA Category II/III.

The scope of Modification F1-95-121 includes the installation of shielding around the DHR equipment to be located on elevation 326 of the Reactor Building. Said shielding will be installed in accordance with QA Category II/III, Seismic Class II design criteria. Shielding installation is not a prerequisite to either system construction/installation or operation. Refer to section 4.5 of this report for additional information regarding shielding installation.

4 Engineering and Design Issues

4.1 Single Failure Considerations and Overall Plant Decay Heat Removal Capability

The total decay heat load, RPV plus SFP, is shown in Table 1. Details of how that table was generated are contained in section 4.10, below. As shown in Table 2, the heat removal capacity of the DHR is significantly greater than all existing plant systems which can remove heat from either the SFP or the RPV (during refueling outages). The ensuing discussion compares the pre- and post-DHR decay heat removal capabilities during three distinct phases of a typical refueling outage (assuming a full core offload was performed) with corresponding differing decay heat load conditions. The consequences of postulated single failures or natural phenomena (i.e., design basis earthquake or high wind event) on overall plant decay heat removal capability during each phase are also evaluated.

The existing plant design provides two primary means of decay heat removal from the SFP during refueling outages; the FPCCS and RHR operation in the fuel pool assist mode. Neither of those systems/configurations are designed to remain functional following a postulated passive failure. The existing design provides one primary means of decay heat removal from the RPV during refueling outages; RHR operation in the SDC mode. RHR SDC is not designed to remain functional following a postulated passive failure. Hence, the DHR design mimics the design characteristics of existing plant (refueling outage) decay heat removal features in ensuring system capability in the nominal configuration can be maintained, with appropriate operator action, following a broad spectrum of postulated active component failures, including loss of the normal system power supply.

4.1.1 Phase 1: From Plant Shutdown to Removal of the SFP Gates

Prior to floodup of the reactor cavity and removal of the SFP gates, the maximum decay heat load which could be placed on the DHR would be the SFP decay heat load. Per reference 4 the typical (pre-outage) SFP heat load is less than 2 X 10^6 BTU/HR. While the system is capable of cooling the SFP (only), that configuration is not a design consideration for the system given its nominal heat removal capacity of 30 X 10^6 BTU/HR at design conditions. As detailed in section 3.1.1 above, the automatic fan control circuitry provides a range of heat removal capability from approximately 2 X 10^6 BTU/HR to approximately 30 X 10^6 BTU/HR at an ambient wet bulb temperature of 73 ^{0}F. Refer to Section 8.1 of this report for evaluation of DHR operation in lieu of FPCC. Although this condition (Phase 1) is essentially not applicable to the DHR, it is instructive to review existing plant decay heat removal capabilities during this phase of a RO.

4.1.1.1 SFP Cooling

The existing plant design provides two means of removing decay heat from fuel in the SFP. The normal means is via operation of the non-safety related Fuel Pool Cooling and Cleanup (FPCC) system. As delineated in the UFSAR, the fuel pool cooling heat removal function is a power generation objective and design basis; that is, it is not a safety-related function. The (original design) heat removal capabilities of the FPCC are approximately 6.3 and 10 X 10^6 BTU/HR in the nominal and maximum configurations, respectively. The FPCC is not powered off of the plant safeguards buses and would be unavailable following a postulated loss of offsite power (LOOP). The FPCC is not seismically designed and would not be available following a postulated design basis seismic event.

Under accident conditions and in the initial period after a full core offload, the safety related Residual Heat Removal (RHR) system, operating in the fuel pool assist configuration, can be used to supplement or substitute for FPCC operation. RHR system components are powered from safety-related sources and would be available following a postulated LOOP. The RHR system is seismically designed and would be available following a postulated seismic event. However, the RHR fuel pool assist piping ties into the FPCC piping, which, like the DHR, is supported to meet the QA Category II/III, Seismic Class II design criteria. Hence, RHR fuel pool cooling assist capability is not assured following a design basis earthquake.

4.1.1.2 RPV Cooling

Prior to installation of the DHR, a postulated loss of SDC could only be mitigated (directly) by restoration of SDC. While such efforts were taking place, RPV and reactor cavity water temperatures would increase, and unless SDC were restored, the reactor cavity water would eventually boil. Installation of the DHR has no bearing on that scenario as it can not remove heat from the RPV until the reactor cavity has been flooded and the SFP gates removed.

4.1.2 Phase 2: SFP Gates Removed and the Total Decay Heat Load Exceeds DHR Capability in the Nominal Configuration

During this phase the decay heat load is assumed to exceed the rejection capacity of DHR operating in the nominal heat removal configuration. Conservatively assuming the DHR were operating at its design condition, i.e., at a wet bulb temperature of 73 ^{0}F, the decay heat load during this phase must exceed 30 X 10^6 BTU/HR. Based on reference 4, such a heat load conservatively corresponds to approximately 108 hours (4.5 days) post-shutdown. As the plant is currently prevented from initiation of core offload for the first 96 hours, post-shutdown, and is limited to transfer of four assemblies per hour from the RPV to the SFP, due to SFP heat removal capability, in the pre-DHR configuration it is highly unlikely a significant amount of fuel (and hence decay heat) would have been transferred to the SFP during this phase of the outage.

4.1.2.1 SFP Cooling

Prior to installation of the DHR, the decay heat load in the SFP was limited (based on the aforementioned limitations on fuel transfer) such that the peak SFP temperature following either discharge of a reload batch (208 assemblies) or a core offload coincident with a postulated single failure in the FPCC would remain less than 150 ^{0}F. Refer to reference 5 for details of that evaluation. Under those conditions, pool heatup was limited by operation of RHR fuel pool assist and FPCC (single pump operation). During phase 2 of the RO, i.e.,during the initial stages of fuel transfer and prior to completion of a core offload, the decay heat load in the SFP would remain within the capabilities of FPCC. If FPCC were not available, RHR fuel pool assist would be initiated. Installation of the DHR in no way detracts from the ability to either restore FPCC following a postulated failure or initiate RHR fuel pool assist. With DHR available, the postulated loss of FPCC under these conditions could be mitigated by simply operating the DHR system in its nominal configuration. Operation of the DHR under these conditions would limit SFP heatup and obviate the need to initiate RHR fuel pool assist.

4.1.2.2 RPV Cooling

Prior to installation of the DHR, a postulated loss of SDC could only be mitigated (directly) by restoration of SDC. It should be noted the SDC configuration is not single failure proof. While such efforts were taking place, RPV and reactor cavity water temperatures would continue to increase, and unless SDC were restored, the reactor cavity would eventually boil. With the DHR installed, and operating in its maximum configuration, natural circulation cooling of fuel assemblies in the RPV would be established and RPV and SFP heatup precluded. Under such conditions, DHR operation in the nominal configuration would significantly decrease the net heat addition to the combined RPV and SFP water volumes, establish natural circulation cooling of fuel assemblies in the RPV, and effectively extend the period of time available for restoration of SDC.

4.1.3 Phase 3: SFP Gates Removed and the Total Decay Heat Removal Load is Less Than DHR Capability in the Nominal Configuration

During this phase the decay heat load is within the rejection capacity of the DHR in the nominal configuration. As noted above, this would occur approximately 4.5 days post-shutdown if the DHR [illegible] operating at a wet bulb temperature of 73 ^{0}F. At this point in a RO the amount of fuel [illegible] from the RPV to the SFP could vary widely. The aforementioned limitations on fuel [illegible]t, see section 4.1.2 above, effectively limit the rate of decay heat transfer to the SFP in the [illegible] DHR plant configuration, with the majority of heat remaining in the RPV.

4.1.3.1 SFP Cooling

SFP cooling issues during phase 3 are identical to those described for phase 2 above with the exception a full core offload would be completed during this phase and the peak SFP water temperature following a postulated FPCC failure would be less than 150 ^{0}F. The existing limitations on fuel movement are important in the pre-DHR configuration as those limitations are required to ensure adequate SFP cooling (the delay in offload initiation and the slow transfer rate serve to both minimize the SFP heat load directly and to maximize the amount of heat removal by SDC). In the post-DHR configuration the extent to which fuel has been transferred to the SFP is irrelevant as the DHR heat removal capability is independent of the location of the fuel. Under these heat loads, the DHR would be operating in the nominal configuration. Given a postulated failure of a DHR component, its redundant counterpart could be placed in service or the "failed" component by-passed and nominal heat removal capability restored. SFP heatup would be limited to the short period of time needed to "realign" the DHR.

4.1.3.2 RPV Cooling

RPV cooling issues during phase 3 are identical to those discussed for phase 2 above, with the exception that the decay heat load would have decreased in proportion to the extent of completion of the offload. The heat load in this configuration would have continued to be picked up by SDC, until core offload was completed, and a postulated loss of SDC could only be mitigated through restoration of SDC. Post DHR installation, a postulated loss of SDC could be mitigated either through restoration of SDC or operation of the DHR in either the nominal or maximum configurations. Given the vast heat removal capacity of the DHR, it would be possible for the plant to withstand the theoretical failures of SDC, FPCC, and a DHR component failure in this configuration while maintaining adequate decay heat removal capability.

As detailed in section 5.2 below, the DHR system is capable of removing decay heat from both the SFP and the RPV. Thus, the ensuing evaluation is valid regardless of the location of the fuel and independent of the status of core offload or reload activities.

4.2 Seismic Design

The DHR is not designed to remain functional following a design basis earthquake. DHR equipment and piping will be supported such that their (functional) failure during a design bases earthquake will not adversely effect safety-related equipment. That being the case, installation and operation at the time of a postulated earthquake would have no adverse effect on plant safety.

4.3 System Power Supply

Electric power for the DHR will be provided from a reliable offsite source (13.2 KV). An underground feeder, consisting of a conduit encased in concrete, will be installed from the "grey shed" in the yard to a 13.2 KV disconnect switch and to a 13.2 KV/480V transformer. The transformer will be located in the yard east of the Railroad and Truck Bay and Gas Treatment Building. Adjacent to the transformer will be a 480V motor control center (MCC). The MCC will distribute power to all DHR components. A system one-line diagram is contained in figure 3.

The DHR electrical supply and distribution system is physically separate from existing safety related and non-safety related electric equipment in the power block.

For scheduled refueling outages, DHR reliability will be enhanced through use of a portable diesel generator which will be directly connected to the DHR 480V MCC. The portable diesel generator will be sized to start and carry DHR loads for operation in the nominal configuration and would be used in the event of loss of the normal 13.2 KV supply. Transfer from the 13.2 KV source to the diesel generator would be accomplished manually.

Adequate lighting will be available in the yard area for access to and operation of the portable diesel and for transfer of the DHR MCC power supply from the 13.2 KV feed to the diesel (at the DHR MCC). The interior security lighting is supplied from an in-plant source and would be unaffected by the loss of the normal 13.2 KV DHR supply. Additional lighting (the equivalent of industrial "life safety" lighting units) will be provided local to the rental diesel skid to ensure adequate illumination to support local operator action. Modification F1-95-121 will provide for lighting local to DHR equipment to be located on the roof of the Railroad and Truck Bay and Standby Gas Treatment Building roof. Those lights will ultimately be powered from the 13.2 KV system and, in the event of loss of the normal 13.2 KV system supply, would not be in service pending reenergization of the MCC from the rental diesel.

4.4 Electrical Separation

All cabling associated with the DHR will be in accordance with applicable plant standards and procedures. All routing will be in accordance with existing plant procedures and standards to ensure the appropriate separation criteria are met.

4.5 Operational Radiation Fields

Radiation fields in the vicinity of DHR components and piping are expected to be the same as the fields experienced outside the FPCC heat exchanger and pumps rooms with FPCC in operation. The primary difference between the FPCC and the DHR in terms of operational radiation fields could be associated with the lack of shielding of the DHR primary pumps, strainer, and heat exchangers (vice the existing shielding of the analogous FPCC components). Radiation fields in the vicinity of the pumps and heat exchangers will be monitored during DHR operation and corrective actions initiated, if necessary. The exact nature of any corrective actions would be dictated by RES personnel in conjunction with Engineering. It should be noted the scope of Modification F1-95-121 includes the installation of shielding around the DHR equipment to be located on elevation 326' of the Reactor Building. Shielding installation is expected to be completed prior to RO12 but is not a prerequisite for system operation.

The above mentioned shielding was not installed prior to or during RO12.

4.6 **Fire Protection Issues**

Installation of the DHR will result in additional combustible loading in the affected plant areas. Fire Protection review of those loadings, as well as review of the DHR for Appendix R considerations, will be performed as part of the Modification process.

4.7 **Wind Loadings on Exterior Components**

Secondary side components to be located outdoors will be evaluated for 90 mph wind conditions. Those wind conditions are cited in the NRC SER for the facility, reference 6.

4.8 **Internal Flooding**

Installation of the DHR will result in new energized piping routes in the Reactor Building. Review of those routes for potential flooding, equipment qualification effects, and area accessibility will be performed as part of the Modification process.

4.9 **Heavy Loads**

Installation of the DHR is expected to take place while the unit is on line and it is imperative all equipment and piping rigging and lifts be performed in accordance with existing plant procedures and standards for movement of heavy loads. Of particular import are rigging and lifts internal to the Reactor Building and the SFP. All rigging and lifting will be performed in accordance with reference nos. 7 and 8.

The rigging and lifting of the cooling tower cells will be performed outside the reactor building over the Railroad and Truck Port Building. The cooling tower cells will not be lifted over safety related equipment. The following requirements will be used when performing these lifts:

1) Slings, shackles, and related rigging will have a factor of safety of 10:1 ultimate capacity to the actual load with the exception of the two redundant rigging slings or cables beneath each cell, which will have a minimum service capacity of 10,000 lbs. each.

2) The spreader beam will be load tested to 150% of the service load (including an impact factor) for a period not less than 10 minutes. Additionally, the critical areas including load bearing welds will be inspected for defects.

3) The truck crane will have sufficient lifting capacity such that the cooling tower cell lift load is not more than 85% of the crane load rating during any configuration during the lift.

4.10 Decay Heat Load Calculations

RPV and SFP decay heat loads throughout the life of the plant have been calculated using conservative methodologies and assumptions. Using the methodology of Branch Technical Position (BTP) ASB 9-2 (Reference 9), an estimate of the end of life decay heat loads versus time after shutdown is provided in Calculation JAF-CALC-MISC-02244 (Reference 4). These heat loads are quite conservative based on the following assumptions used:

1) Operation at the power uprate level of 2536 MWt is assumed (versus the present operating power level of 2436 MWt). The worst case heat input case considered corresponds to a high-energy 24 month cycle fuel load in the RPV (or the SFP).

2) The original calculation conservatively considered an SFP capacity of 2861 bundles, or sixty-four (64) assemblies above the current licensed capacity of 2797 assemblies. The full core offload is 560 bundles. The calculation was revised subsequent to RO12 to reflect the proposed increase in SFP storage capacity to 3247 assemblies.

3) An end of life, full spent fuel pool less a full core offload capability (2861 - 560 = 2301), was assumed in the original calculation, i.e., prior to RO12. The calculation was revised to reflect the aforementioned proposed increase in SFP storage capacity. (Note: If the SFP and RPV were both assumed full, the SFP heat load would not significantly increase, since the 560 empty spaces in the SFP would be filled with the oldest fuel in inventory. Using the oldest fuel would contribute less than 15% to the SFP heat load.)

4) The energy of the fuel batches discharged to the pool is maximized. Lifetime plant capacity between the end of Cycle 12 and the end of life is assumed to be 90%. Fuel cycles are assumed to be 24 months with a 40-day refueling, and 95% capacity factor from startup to shutdown.

The above assumptions were used to generate the decay heat versus time curve used to size the DHR system (nominal configuration). An additional margin of 10% was not added since the results are used to size the DHR heat removal capacity, and an additional 10% over sizing is deemed not necessary because the DHR will be the sole decay heat sink only when the heat load has decreased to within the capacity of the system (operating in the nominal configuration). (Note: The ASB BTP specifies application of margin to the decay heat curve when used as input to safety-related analyses, i.e., 10 CFR 50.46 ECCS performance evaluations).

An additional set of calculations have been performed to estimate maximum decay heat. Those calculations are summarized in revision 1 to JAF-CALC-MISC-02244. As expected, the more conservative analytical model predicts higher heat loads for assemblies in either the RPV or the SFP. Using those results, the estimated time during a typical outage when the DHR (nominal configuration) was to become capable of handling the combined RPV and SFP decay heat loads would be approximately 5.8 days, or thirty one (31) hours longer than was predicted by the original, more realistic, design calculation. The differences between the two calculated results have no bearing on the DHR as system operability during an outage is based on actual test results and not on calculated decay heat loads.

NRC IE Information Notice No. 96-39, "Estimates of Decay Heat Using ANS 5.1 Decay Heat Standard May Vary Significantly", Reference 23, has been reviewed and found to be not applicable to the analyses performed in support of the DHR. The Authority's review is documented in Reference 24.

Note: Throughout this report, DHR heat removal capacity is expressed in terms of approximate number of days, or hours, post-shutdown. These times are based on conservative analysis of the worst-case design heat load condition and are provided for comparative purposes only. The actual time equivalents (as experienced in the plant) would be less than those predicted by such calculations, as can be seen from the conservative assumptions presented above to derive the decay heat values. The calculated values of the SFP plus RPV decay heat, versus time after shutdown, for the time periods relevant to this evaluation, are given in Table 1.

Based on Calculation JAF-CALC-MISC-02244, the DHR system is sized to provide a design maximum heat removal capability of 45×10^6 BTU/HR when using one primary pump through both heat exchangers and using both secondary side pumps and both pair of single-celled cooling towers. In the design normal/nominal heat removal configuration, the DHR system heat removal capability is 30×10^6 BTU/HR at system design conditions. These capabilities far exceed the capabilities of the existing plant systems/configurations capable of cooling both the SFP and the RPV (with the exception of RHR SDC). Thus, installation and operation of the DHR system represents a significant enhancement in the decay heat removal capability of the facility.

The relative decay heat removal capabilities of existing systems and the DHR are summarized in Table 2 . The detailed inputs to the respective heat removal capacity calculations are not identical, since these capacities were generated by different analyses and use different initial conditions and assumptions. In all cases, however, the design inputs for the DHR sizing are the most conservative.

Figure 4, attached, presents the decay heat loads, and the heat removal capabilities available from the DHR system the RHR SDC or RHR assist modes, and the FPC system, versus time after shutdown.

The only plant feature capable of removing a greater amount of decay heat is RHR SDC. As can be seen on Figure 4, installation of the DHR system does not eliminate the need for SDC operation at the beginning of an outage.

5 Nuclear Safety Considerations

5.1 Maintenance of SFP Water Level Following a Postulated Breach of System Pressure Boundary

The DHR system suction and discharge piping in the vicinity of the SFP include holes to preclude siphoning (either gravity drainage or pump down) of the SFP in the event of a breech in DHR primary loop pressure boundary. Those holes are functionally equivalent to the vacuum breakers provided on the existing FPCC. The elevation of the holes in the DHR sparger pipes will be such that adequate inventory would be maintained in the SFP following a postulated breech of DHR primary loop pressure boundary.

The potential for loss of refueling cavity inventory due to postulated failure of the of the reactor cavity seal has previously been evaluated for JAF as part of our response to INPO Significant Operating Experience Report (SOER) 85-1 (reference 10). Installation and operation of the DHR has no bearing on the SOER evaluation in terms of the design of the cavity seal and the potential for catastrophic failure. As detailed in the SOER response, a design review was performed for JAF which concluded gross failure of the seal was not a credible event. The review went on to address associated issues such as the installation of nozzle dams, etc. The results of those reviews are not affected by DHR operation. Outage risk management considerations specific to the situation where DHR was being utilized in lieu of RHR are defined in section 8.3 below.

5.2 Natural Circulation Cooling of Fuel Assemblies in the RPV

Thermal hydraulic calculations performed by General Electric (Reference 11) and NYPA Calculation JAF-CALC-RHR-02380 (Reference 12) confirm the ability of the DHR to remove the decay heat from fuel assemblies located in the RPV; that is, prior to core offload and subsequent to reload. The calculations assume the RPV head removed, the reactor cavity to be flooded up, and the SFP gates to be removed. Under such a condition, i.e., prior to core offload, decay heat from the fuel in the RPV would be removed by natural circulation. The analyses also confirm natural circulation cooling would continue throughout the offload process. These calculations confirm the establishment of natural circulation currents in the RPV and reactor cavity whenever the DHR is operated with the SFP gates removed and the cavity flooded to refueling level.

The analyses considered the following factors:

1) operation of the DHR in its design maximum heat removal mode or its nominal heat removal mode

2) the time after shutdown at which the core is offloaded (the calculations assume offload into the SFP commences at 72 hours and is complete at 108 hours post-shutdown)

3) the location of the DHR discharge pipe in the SFP, including the theoretical case where the DHR suction and discharge pipes are in close proximity to each other and near the surface of the pool.

For these analyses, the initial bulk SFP water temperature is assumed to be at 114 ^{0}F. The results of those calculations show an initial increase in RPV water temperature, essentially core average temperature, upon termination of RHR SDC and forced flow through the core. The magnitude of temperature increase is conservatively predicted to be on the order of 15-20 ^{0}F. During this period of temperature increase, the water volumes in the RPV, the reactor cavity, and the SFP are approaching a quasi-steady state equilibrium condition. Approximately one hour after SDC termination, natural circulation flow patterns are established and RPV water temperature begins to decay exponentially and approaches the SFP water temperature.

The peak core exit temperature predicted by either the GE or NYPA calculations is 138 ^{0}F. The peak water surface temperature is predicted for the theoretical case where the DHR suction and discharge pipes are in close proximity to each other and near the pool surface. The peak value for that case is 133 ^{0}F. The maximum predicted bulk surface temperature of the SFP/RPV/reactor cavity water volume for all other cases analyzed is 118 ^{0}F.

Sensitivity analyses indicate the surface temperature of the combined SFP- reactor cavity water volume is relatively insensitive to:

1) the fraction of fuel offloaded from the RPV,

2) DHR operation at design maximum heat removal capacity (45 X 10^6 BTU/hr) versus nominal heat removal capacity (single train operation removing up to 30 X 10^6 BTU/hr), and

3) the approximate time, post-shutdown, of DHR initiation, 72 hours vice 108 hours, assuming no other method of decay heat removal.

Details of natural circulation verification to be performed during RO12 are given in section 7.1 below.

General Electric has analyzed the heat transport from the core (RPV) to the SFP for Plant Hatch, Hope Creek, and the J.A. Fitzpatrick plants (Reference 13). A review of the geometry of the RPV/SFP/reactor cavity at these plants was performed. The parameters included the distance between the core centerline to fuel pool gate, the elevation of the bottom of the canal versus the RPV flange, the elevation difference of the flange versus the refueling floor deck, the elevation difference between the bottom of the canal and the bottom of the pool, and the fuel pool gate width. The review concluded that there were no significant differences between the three plants that would affect natural circulation flow between the core and the refueling pool. Therefore, there is no technical requirement to test the natural circulation capability; that is, one is merely proving that hot water rises and cold water sinks.

During performance testing of the DHR in RO12 the RWCU and CRD systems were in service and questions were raised by the Plant Operating Review Committee (PORC) and NRC personnel as to the effects, if any, of CRD and RWCU on DHR operability. The questions centered on three issues;

(1) the effect of cold water addition by CRD,

(2) the effect of heat removal via the RWCU system, and

(3) the effect of forced flow (both CRD and RWCU) on natural circulation cooling of irradiated fuel in the RPV

Responses to PORC on those issues were provided via reference nos. 31 and 32. Said references were also discussed with NRC personnel. Subsequent to RO12 the natural circulation cooling analysis was revised to (analytically) reflect operation of CRD and RWCU. Based on the results of the revised analysis it was concluded CRD and RWCU flows would have no significant effect on natural circulation flow characteristics in the RPV, reactor cavity, and SFP.

5.3 Potential for Excessive Cooling of the SFP

The heat removal capability of the DHR is vastly greater than the decay heat load of the SFP during non-refueling outage conditions and nominal heat removal capacity exceeds the combined RPV and SFP decay heat loads approximately 4.5 days post-shutdown. As detailed in section 3.1.1 above, the automatic fan control circuitry provides a range of heat removal capability from approximately 2 X 10^6 to approximately 30 X 10^6 BTU/HR when the system is in the nominal configuration at an ambient wet bulb temperature of 73 ^{0}F. Thus, the potential for overcooling the SFP is low. However, there is a finite potential for such an occurrence given hypothetical failure of the fan control circuitry. The lower temperature limit on SFP temperature is a plant safety limit derived from criticality analyses of spent fuel stored in specific rack designs. The lower temperature limit for the RPV is based on metal brittle fracture analyses.

Operation of the DHR is acceptable provided the appropriate administrative controls and procedures are in place to ensure plant operators are aware of the potential for overcooling and have been properly trained in mitigating actions. Under any condition or configuration, manual trip of operating primary side pumps at the DHR control panel would terminate any possible cool down.

Subsequent to RO12 additional analyses were performed to support a decrease in the minimum allowable SFP bulk temperature from 68 ^{0}F to 32 ^{0}F. The additional analyses are included in reference no. 30.

5.4 Technical Specification and Design Bases Document Review

The following pertinent Technical Specification sections were reviewed to perform this report.

3/4.0	Applicability (and Bases)
3/4.2	Instrumentation (and Bases)
3/4.7	Containment Systems (and Bases)
3/4.9	Auxiliary Electrical Systems (and Bases)
3/4.10.C	Spent Fuel Storage Pool Water Level (and Bases)
3/4.11	Additional Safety Related Plant Capabilities (and Bases)
6.0	Administrative Controls, including Sections 6.5, Review and Audit, and 6.8, Procedures and Programs

Based on review of the above sections, it has been determined a change to the Technical Specifications is not required.

Installation of the DHR does require a change to the Design Basis Documents. The Design Basis Document for the Residual Heat Removal System, 010 (Reference 14), requires a discussion of the use of the DHR system.

5.5. UFSAR Review

UFSAR Sections 9.3 and 9.4 (as a minimum) will be revised to clarify and update the decay heat loads assumed, the various heat removal assumptions, and the various modes of operation which are available to the operators with the addition of the DHR system. Clarification is required due to the addition of the DHR system, but also due to the various decay heat removal assumptions used in previous submittal, the number of fuel assemblies assumed in the SFP, the discussions of RHR assist as the only method, etc. Also, the results of the thermal analyses which support installation and operation of the DHR system show DHR operation to be independent of fuel location. Hence, the UFSAR will be revised to reflect the fact existing limitations on fuel movement, a minimum delay of 96 hours post-shutdown prior to initiation of fuel movement and a maximum fuel transfer rate of four assemblies per hour into the SFP, would not be applicable when the equipment necessary to support DHR operation in the maximum heat removal configuration was available at the time of initial fuel movement. Said limitations would remain in effect if the DHR (maximum heat removal configuration) were not available during a refueling outage.

A draft UFSAR revision will be prepared as part of Modification Package number F1-95-121, "Decay Heat Removal System, Project CJ311O".

The following UFSAR Sections were reviewed to perform this SE, and to determine whether a change to the UFSAR is required:

Section 1.2	Definitions
Section 1.3	Methods of Technical Presentation
Section 1.4	Classification of BWR Systems Criteria, and Requirements for Safety Evaluation
Section 7.6	Refueling Interlocks
Section 7.12	Process Radiation Monitoring System
Section 7.13	Area Radiation Monitoring System
Section 8.6	Emergency AC Power System
Section 9.3	Spent Fuel Storage
Section 9.4	Fuel Pool Cooling and Cleanup System
Section 9.9	Heating, Ventilation, and Air Conditioning Systems, including Section 9.9.3.3, Reactor Building Ventilation System
Section 12.	Classification of Structures and Equipment
Section 12.3	Description of Principal Structures
Section 12.4	Structural Loading Conditions
Section 12.6	Analysis of Spent Fuel Storage Pool
Section 13.8	Plant Procedures
Section 14.5	Analysis of Abnormal Operational Transients and Reactor Vessel Overpressure; including Section 14.5.8, Event Resulting In A Core Coolant Temperature Increase (loss of RHR shutdown cooling)
Section 14.6	Analysis of Design Basis Accidents
Chapter 16	Appendices: Section 16.5, Pressure Integrity of Piping and Equipment Pressure Parts; and Section 16.6, Conformance to AEC Design Criteria
Chapter 17	Quality Assurance Program: Appendix 17.2B, Conformance with NRC Regulatory Guides; and Appendix 17.2C, Plant Administrative Procedures General List

The DHR system plays no role in either prevention or mitigation of a fuel handling accident (FHA) and the FHA consequence analysis contained in the UFSAR is unaffected by DHR installation and operation. A postulated FHA would have no direct effect on DHR system operation, i.e., if the DHR were in operation at the time of a postulated FHA the (DHR) system would continue to operate and remove decay heat. The DHR control panel will be located on elevation 326 of the Reactor Building in the immediate vicinity of the existing FPCC control panel and, in the unlikely event of a FHA concurrent with an independent failure of a DHR component, personnel access to that area of the plant would be unchanged from the pre-DHR plant configuration.

6. System Installation and Modification Acceptance Testing

As mentioned above, the system is expected to be installed and tested prior to RO12 and the unit is expected to be on line during that period. The appropriate administrative controls and procedures are in place to ensure system installation does not adversely effect plant operation or plant safety.

Acceptance testing will be performed with the system isolated from the SFP, i.e., spool pieces will be used local to the SFP to provide a primary side flow path which does not involve circulation of radioactive fluid to and from the SFP.

7. Initial Thermal Performance Testing of the DHR System Decay Heat Removal Capability

Modification acceptance testing of the DHR will be performed prior to RO12. However, it is not meaningful to perform a thermal performance test of the DHR with the system aligned for SFP cooling (only), i.e., prior to reactor cavity floodup and removal of the SFP gates, due to the limited heat load. Therefore, performance testing of the system will be performed during RO12. Thermal performance testing will serve four distinct purposes:

(1) during RO12 the initial thermal performance test will verify the system is capable of performing as designed,

(2) during RO12 and each subsequent outage the DHR is placed in service, thermal performance data will be used to estimate the decay heat load,

(3) comparison of thermal performance data from outage to outage will be used to monitor for indication of DHR degradation, and

(4) during RO12 and each and any subsequent outage when RHR is to be removed from service, DHR thermal performance data will be used as part of the outage risk assessment process for the determination of when RHR can safely be removed from service.

7.1 Thermal Performance Testing During RO12

The DHR system will be placed in service and FPCC and SDC secured after the SFP gates are removed. With SDC secured, but available, the ability of the DHR to remove the combined RPV and SFP decay heat load will be functionally verified by operating the system and verifying a negative trend in DHR primary side suction temperature. The combined SFP and RPV heat loads at that point are expected to be within the capability of the DHR operating in the nominal heat removal configuration.

Detailed engineering data obtained during that (and subsequent periods) will be used to evaluate DHR thermal performance. However, the functional test of the DHR will be simply to maintain or decrease pool water temperature when RHR and FPC are not in service.

7.2 Verification of Natural Circulation During Initial System Operation (RO12)

The heat removal capability of the DHR is independent of location of the fuel and hence is not chronologically tied to the initiation of core offload activities. However, it is expected - but not required - that both the maximum and nominal heat removal configurations of the DHR system will be tested prior to initiation of offload. Assuming such to be the case, the establishment of natural circulation cooling of fuel in the RPV will be verified by observation of a decreasing trend in SFP water temperature with RHR SDC not in operation.

7.3 Summary of DHR Performance During RO12

From a heat removal perspective, the DHR functioned in admirable fashion during RO12. Natural circulation flow patterns were established quickly subsequent to removal of RHR from service and the overall trend of SFP and RPV temperatures were as expected. Refer to figure no. 5, and the accompanying data table, for time-temperature trends observed and measured at the start of RO12. The automatic fan control features allowed for system operation to automatic "react" to the decrease in decay heat load throughout the outage. Operation of the DHR resulted in decreased SFP and reactor cavity temperatures which enhanced fuel handling operations and personnel comfort. DHR operation did not result in any fuel handling problems due to thermal currents from the reactor cavity into the SFP.

Throughout the outage, operational problems were experienced with the strainer installed in the DHR primary loop. The strainer was bypassed, as designed, to allow for continued system operation. Resolution of strainer design issues is ongoing. Throughout the outage,the system was operated without the benefit of additional (permanent) shielding. While dose levels were acceptable in the area, Elevation 326 of the RB, the need to install the permanent shielding was apparent. As of this writing, installation of the permanent shielding provided for in MOD F1-95-121 is planned to be completed prior to RO13. The contribution of the aforementioned strainer problems to observed radiation fields is not quantifiable, but clearly, resolution of the strainer difficulties would be beneficial from a long-term ALARA perspective.

The calculated heat removal capability of the system (in the nominal heat removal configuration) while adequate to fulfill decay heat management requirements, was less than expected based on system design considerations. Specifically, while the system was shown to be capable of removing 30 X 10^6 BTU/HR at the conditions which existed at the time of the outage, extrapolition of those results to the design condition did not conclusively show the system capable of meeting that design requirement. Of the multitude of parameters which effect DHR heat removal capability, the two most critical to heat rejection capability at any given point in time, assuming proper operation of mechanical equipment, are outside wet bulb temperature and SFP fluid temperature. During RO12 the outside conditions were less restrictive (cooler) than the design condition while the SFP fluid temperature was deliberately lower (initial temperature on the order of 115 ^{0}F vice the design point of 125 ^{0}F). Resolution of this apparent discrepancy is ongoing at this time. It should be noted this apparent discrepancy has no bearing on the safety aspects of DHR operation, including the decision to render RHR unavailable under specified conditions, since DHR operability during RO12 was determined by actual test at the conditions which existed. Similar tests will be performed during subsequent outages when DHR operation is intended.

8. System Operation and Testing

The following sections discuss, in general terms, the various possible plant conditions which could exist when the DHR might be placed in service. Detailed operating, test, maintenance, and surveillance procedures will be developed, as appropriate, prior to system operation.

8.1 **DHR Operation with the Unit on line**

The DHR can be used in lieu of FPCC to cool the SFP with the unit on line; the sole safety concern being the potential for overcooling which is addressed in section 5.3 above.

There is no technical requirement to maintain a backup diesel generator for system operation in this condition as the SFP decay heat load could be handled using the RHR system in the fuel pool assist mode. Details of the use of RHR for SFP cooling given the postulated unavailability of the FPCCS are contained in Reference 19.

8.2 **DHR Operation During Refueling Outages - SFP Gates Installed**

System operation in this configuration is acceptable. At the start of the outage, this operating condition is identical to use of the DHR in lieu of FPCC with the unit on line. During ROs (when the SFP gates are installed) there are two possible situations during which the DHR could be used.

The first condition would correspond to a hypothetical situation wherein core offload had been completed and for some reason the SFP gates were installed, i.e., to facilitate in vessel inspections or repairs. The DHR could be operated in the nominal configuration in lieu of FPCC and RHR fuel pool assist.

The second condition would occur at the end of each outage after core reload had been completed and in advance of vessel reassembly activities. The DHR could be operated in this configuration in lieu of FPCC and RHR fuel pool assist.

8.3 **DHR Operation During Refueling Outages - SFP Gates Removed**

Operation of the DHR in this configuration is the design configuration for the system. As detailed in section 4.1 above, operation of the DHR in the nominal configuration (beginning approximately 4.5 days post-shutdown) eliminates the need for FPCC and RHR (SDC and fuel pool assist) operation while maintaining full redundancy for decay heat removal.

The existing JAF Technical Specifications do not require RHR SDC operability when the reactor cavity is flooded. However, the Standard Technical Specifications (STS), reference 15, would require one train of SDC to be available and operating as long as irradiated fuel were in the RPV. The JAF Technical Specifications do not mention the DHR. However, the "REQUIRED ACTION" statement of STS 3.9.8 states: "Verify an alternate method of decay heat removal is available" if RHR SDC were not available. The bases for that STS section reads in part as follows: " ...the volume of water above the RPV flange provides adequate heat removal capability to remove decay heat from the core" and goes on to recognize the ability of other plant systems, i.e., Reactor Water Cleanup, to handle the RPV decay heat at some point in the outage.

Based on figure 4, the total (SFP and RPV) decay heat load would decrease below the capacity of the DHR in the nominal configuration (at the design wet bulb temperature) approximately 4.5 days post-shutdown. From that point on in an outage, assuming the cavity flooded and the SFP gates removed, the DHR would be single (active) failure proof, with operator action, for decay heat removal regardless of the status of refueling activities. The unavailability of RHR SDC at that point in an outage would, under the most conservative interpretation, be tantamount to being in the STS "ACTION" statement. In reality, DHR availability at that time would provide redundant means of RPV (and SFP) decay heat removal. It is more reasonable to consider RHR SDC and the DHR (nominal configuration) to be redundant means of decay heat removal when flooded up with the SFP gates removed. By administratively requiring SDC to remain operable until the decay heat load was within the capability of DHR in the nominal configuration, plant management will ensure the decay heat removal function was single (active) failure proof throughout the outage. During the initial part of an outage, i.e., beginning approximately 1.5 days post-shutdown until approximately 4.5 days post-shutdown, RHR SDC and DHR (maximum heat removal configuration) would be considered redundant (provided the cavity were flooded and the SFP gates removed). Once the total decay heat load has been shown to be within the capability of DHR in the nominal configuration, RHR SDC and DHR (with all redundant equipment operable) should be considered as three methods of decay heat removal. It should be noted the detailed analyses presented in section 5.2 of this report, verify the ability to 'monitor RPV temperature' (in the STS sense of the phrase) using temperature instrumentation on the primary side inlet to the DHR heat exchanger.

In order to ensure the decay heat removal function were single (active) failure proof at the time SDC were removed from service, it is necessary to ensure the inboard Main Steam Isolation Valves (MSIVs) are intact whenever irradiated fuel is in the RPV and the DHR is being used for removal of RPV decay heat unless it can be shown a postulated single active failure in the main steam line plug inflatable seal(s) or the associated pneumatic supply would not render the DHR inoperable. The bases for that limitation are as follows. During normal refueling operations, the main steam line plugs serve as part of the RPV pressure boundary and the inboard MSIVs provide a secondary boundary. As documented in JAF-SE-96-011, Rev.1 , "Moving Fuel While the MSIVs Are Being Worked With Main Steam Line Plugs Installed", Reference 25, the postulated failure of the pneumatic supply to the seals would result in a loss of cavity inventory. Assuming operator action to initiate makeup via a core spray pump were initiated ten minutes into the event, the water level would decrease until the leakage rate was equal to the makeup rate via the core spray pump. The equilibrium water level under such conditions would be approximately three feet lower than the initial level. Such a decrease in level would render both the FPCC and the DHR unavailable due to loss of suction. If such an event were to occur with RHR SDC cooling available, then the basic mitigation scheme would be as follows; maintain core spray pump operation, reinstall the SFP gates, run SDC for RPV decay heat removal, reestablish SFP level, and run either FPCC or RHR fuel pool assist (decay heat load dependent) for SFP cooling. If such an event were to occur with the DHR in service and RHR not available, then the above mitigation scheme would not be valid as the reinstallation of the SFP gates, while necessary to reestablish suction for the DHR would isolate the RPV from the SFP and forced cooling of irradiated assemblies in the RPV would not be available. While not an immediate safety concern due to the time which would be available to affect repairs prior to cavity inventory heatup. the NSE for operation of the DHR, JAF-SE-96-042, Reference 28, specifically requires redundancy in decay heat removal capability be verified prior to the removal from service of RHR SDC.

Additional measures to maintain key spare DHR components onsite would provide an even greater measure of security and further enhance the "reliability" of the decay heat removal function and reduce the potential for loss of decay heat removal events.

8.4 Pre-outage System Testing

Modification acceptance testing of the DHR is expected to be completed prior to RO12 and initial thermal performance testing is expected to be completed during the early stages of RO12 (just after floodup and removal of the SFP gates). Prior to each subsequent refueling outage the DHR will be tested using a subset of the initial modification acceptance test procedure(s). The scope of that testing will include testing of the portable diesel generator to be used during the particular outage. As noted in section 7 above, it is not meaningful to perform a thermal performance test of the system aligned for SFP cooling (only) due to the limited available heat load. Hence, thermal performance testing will not be performed prior to subsequent refueling outages. However, the DHR operating procedure will require thermal performance data be obtained during initial system operation during each outage. The procedure will also include the estimation of total (SFP and RPV) decay heat load based on the data collected. System performance data from outage to outage will be used to monitor for signs of DHR performance degradation.

9. Additional Supporting Analyses

9.1 Evaluation of Postulated Single Active Failures of DHR Components, and Prospective Mitigating Actions with the System Operating in the Nominal heat Removal Configuration

As noted above, the DHR is designed to withstand a single active failure while maintaining, with operator action, a heat removal capability of 30 X 10^6 BTU/HR. An assessment of those potential failures, and associated mitigating actions, follows.

9.1.1 Postulated Active Failure of a Mechanical Component

The normal configuration of the DHR system, a single primary pump and heat exchanger with heat removed by a single secondary pump and single, two-cell, cooling tower package would be in service. Therefore, single active failure of any single mechanical component in the primary or secondary side is mitigated by operation of the backup component. Moreover, if one cell, basin, or fan of a cooling tower package fails then the backup components using one of the two backup tower cells may be placed into service. In addition, the design allows for the operation of either primary or secondary pump operating "through" either heat exchanger. These design features provide improved capability and flexibility in comparison to the FPCC system, wherein a failure of one 50% capacity heat exchanger results in immediate loss of 50% of the heat removal capacity.

The primary side strainer is the single primary side component which does not have a backup. This strainer is provided for ALARA purposes; that is, to preclude excessive buildup of radioactive particles in the heat exchangers. A loss of the strainer by plugging or fouling is mitigated by bypassing the strainer and continuing with the DHR heat removal function. The normal purity and clarity of the SFP water, from continuing use of the SFP filtration system prior to the use of the DHR system, provides assurance that the blockage of the strainer is a low probability event. Moreover, it provides assurance that when the DHR system is placed into service the SFP water will not contain significant quantities of particles which could result in a bypass of the strainer.

The single secondary side component which does not have a backup is the pressure control valve, 32PCV-100, which maintains the secondary side heat exchanger exit pressure greater than the primary side heat exchanger inlet pressure. In the event of a failure, the PCV would be isolated and the manual bypass valve opened. PCV failure might require the operator to restart a secondary side pump. Assuming the initially operating secondary pump tripped, the operator would then start a primary side pump (since the low secondary-to-primary psid would trip the primary pump). These actions restore the decay heat removal capability of the DHR system.

Based on the above, the provision of a backup component to mitigate the failure of a single mechanical component, (or manual action to bypass a failed component) provides a reliable means of removing decay heat from the SFP and/or the RPV. From a mechanical/hydraulic/heat removal perspective, therefore, the DHR system can supplement or replace the FPCC system and provide a system that is more tolerant of a single mechanical failure.

Postulated events or equipment malfunctions which would cause an operating secondary loop pump(s) to trip would also result in automatic closure of the PCV. Since the PCV is at a higher elevation than the cooling towers, a portion of the downstream piping would gravity drain following PCV closure. This was a known condition and was addressed during system design. During modification acceptance testing prior to RO12 was observed sufficient time was available to refill the drained portion of the secondary loop while still fulfilling the "design" objective of being capable of rapidly returning the system to service.

9.1.2. Postulated Active Failure of an Electrical, Instrumentation, and Control Component

The DHR system power is normally supplied from a plant 13.2 KV switchgear J02 to a 13.2 KV-480V transformer outdoor power center. The power center contains an oil-filled transformer to step down the voltage and a motor control center (MCC) which distributes the power to the motors, heaters, and other loads. The MCC contains fused combination starters and contactors and a 120/208 V distribution transformer and panel. The equipment is NEMA grade, sized to handle the full operating current of the loads and available short circuit of the power system. Cables are per ICEA standards, installed in tray and conduit. The equipment is grounded to the plant grounding system per NEC and IEEE standards. Lighting is provided at the power center and at the package cooling tower areas. The DHR system electrical power supplies and distribution system are independent of existing plant systems, both safety-related and non-safety related. Thus, a failure of a DHR electrical system or component cannot affect safety-related electrical equipment or components. During refueling outages, a truck-mounted diesel generator is provided near the DHR system 13.2 KV-480 V transformer to supply emergency power should the normal power supply fail. Thus, for a loss of offsite power, the DHR system is provided with its own dedicated backup power supply, which is also available in the event of a failure of the dedicated 13.2 KV power supply. As indicated on the one-line diagram, fused contactors are provided to each individual load such that failure of one contactor or load does not affect the other loads. Therefore, the loss of one load is enveloped by the considerations of single active failure of mechanical components discussed above.

The DHR system will be manually controlled locally from the control panel. Differential pressure instrumentation across the primary loop strainer is provided to initiate the automatic backwash feature. Differential pressure between the primary and secondary sides of the DHR heat exchangers is also monitored and features provided in the design automatically trip the running DHR primary pump(s) if the sensed pressure differential decreases below 10 psid. A common trouble alarm is provided for the DHR system at the FPC local panel on elevation 326'-9" which also alarms the FPC annunciator in the Control Room. The common trouble alarm indicates any of the following conditions:

1) low flow in the primary system

2) low flow in the secondary system

3) primary loop pump trip

4) cooling tower fan high vibration

5) cooling tower basin low water level

6) transformer winding temperature - high

Postulated malfunction of a secondary-to-primary differential pressure monitoring circuit component would result in the automatic trip of the operating primary loop pump. That circuit can be bypassed at the DHR control panel and a primary pump started (also from the control panel) to restore the system. Differential pressure monitoring would be performed manually using local indicators pending completion of equipment repairs.

A single failure of an I&C component therefore does not preclude the DHR heat removal function, and sufficient redundancy and alarms are provided to give ample time to the operators to effect required repairs.

9.1.3 Evaluation of Potential Mitigating Actions

The mitigating actions necessary to restore DHR operation in the nominal configuration following a postulated single active failure or loss of normal power supply are straightforward and, with the exception of the use of a backup source of secondary side makeup, involve only DHR components. Adequate diagnostic information is available to the plant operators, and the appropriate DHR operating procedures(s) will be in place, to ensure that the DHR system nominal heat removal operation can be restored in a timely manner.

As additional measure, NYPA will maintain selected spare DHR system components available, including spare primary and secondary side pumps, on site during refueling outages. In doing so, and in having the corresponding implementing procedures in place, the time to effect repair and restore DHR redundancy is minimized.

9.1.4 Time Frame for the Implementation of Mitigating Actions

Early in an outage, when the combined RPV and SFP decay heat load might exceed the removal capability of a single DHR train, NYPA will require a RHR SDC train remain OPERABLE. Under those conditions, a postulated DHR failure (of either of the two trains or loss of power supply) could be mitigated by initiation of SDC and, if FPCC is unavailable, RHR fuel pool assist. Such actions are already defined in plant procedures and can be accomplished relatively quickly.

Although not a licensing basis consideration, if RHR is unavailable in the condition where the decay heat load is less than or equal to the capacity of a single DHR train, the parameter chosen herein to assess recovery action is the (minimum) time before boiling occurs in either the SFP or the combined SFP/reactor cavity volume. That time is conservatively calculated as approximately 8.1 hours (per reference 16). Any of the mitigating actions described above (following a postulated single active failure of a DHR component or loss of the normal DHR power supply) can easily be accomplished in significantly less time and 30 X 10^6 BTU/HR heat removal capability restored.

The proposed increase in SFP storage capacity would result in an increase in SFP decay heat (prior to outage initiation) from approximately 1.84 X 10^6 BTU/HR to approximately 1.98 X 10^6 BTU/HR. Such an increase in heat load would tend to cause a decrease in the estimated time-to-boil. Such an increase in SFP heat load is small during non-outage conditions, 0.14 X 10^6 BTU/HR or approximately 7.6% of the previous estimate, and has no bearing on the DHR. The increase would be insignificant relative to the DHR during outage conditions. For example, during an outage which would involve a full core offload, the maximum combined RPV and SFP decay heat load which could exist when the DHR was the sole means of heat removal would be 30 X 10^6 BTU/HR (at DHR design conditions). The estimated change in nominal (non-outage) SFP heat load would represent less than one half of one percent (< 0.5%) of the applied heat load under the stated conditions.

9.2 Estimated Time To Boiling

Calculation JAF-CALC-MISC-02373 (Reference 16) was performed to determine the minimum time to reach boiling conditions in the SFP assuming a full core offload is initiated 2.5 days into the outage and completed by day 5 after shutdown (that is, full core offload in approximately 2.5 days). This conservative assumption is made to maximize the SFP decay heat load at the point in time DHR (and any other methods of decay heat removal) are assumed to become unavailable. The core offload beginning at 2.5 days, and the resultant increase in the decay heat of the SFP, are shown graphically on Figure 4. The reactor cavity is assumed flooded up, and the SFP/RPV/reactor cavity water volumes are interconnected since the fuel pool gates are removed. Under these conditions, the calculated worst-case minimum time to boil in the SFP is approximately 8.1 hours. The maximum SFP heatup rate as fuel is being moved into the SFP over 2.5 days is 12.11F/hr which is concurrent with the minimum time to boil (on day 5 after shutdown). The SFP boiloff rate is calculated to be about 28,588 pounds per hour, and the SFP makeup rate under these conditions is calculated to be about 60 gpm required (based on the revised decay heat calculations contained in revsion no. 1 to reference no. 4). Besides being based on the conservative decay heat loads previously described, this calculation does not account for any heat absorption into the SFP steel liner or concrete walls, nor does it account for any other heat sinks which would delay the SFP heat-up. This time to boil is of interest because it is a very long time for accident mitigation purposes. A time frame of eight or more hours allows ample time to recognize the problem, plan mitigative measures, and carry out those plans. Ample water supplies exist on the nuclear plant site, and Lake Ontario is an ultimate water source if need be. The 200,000 gallon Condensate Storage Tanks are seismic Class I and will survive an earthquake without inventory loss (below grade portions of the tanks), and one tank provides ample capacity for fuel pool makeup. The Fire Protection System, with two diesel-driven fire pumps, is another source of water supply and/or pumping capability, notwithstanding the numerous pumps in the plant. Fuel pool boiling is therefore not a significant safety concern, and, as the next section demonstrates, is not a concern for the public health and safety.

As discussed in section 9.2 above, the proposed increase in SFP storage capacity would result in slight but insignificant decreases in the estimated time-boil under outage and non-outage conditions. However, the magnitude of the increase in SFP heat load is so small as to have no bearing on this evaluation. Note the dose consequence analyses discussed in section 9.3 below are based on the conservative assumption the SFP would begin to boil upon loss of forced cooling, i.e., no credit was taken for the time to heat the pool inventory to boiling.

9.3 Radiological Consequences of Boiling

The licensing basis for JAFNPP does not require performance of either a SFP analysis of the time to reach boiling, or a radiological assessment of the results of a protracted loss of SFP cooling. Nevertheless, these analyses have been performed in support of the DHR system installation and operation. Once again, very conservative assumptions are made; in this case, it is assumed that makeup water is supplied to the SFP to maintain the SFP boiloff, but restoration of SFP cooling is not assumed, i.e., no credit was taken for heat removal due to feed and bleed (steaming) of the pool.

Conservative calculations were performed (Reference 17) to determine site boundary, low population zone (LPZ) and control room doses following a loss of all decay heat removal. The calculation model considers the combined SFP/RPV/reactor cavity water volumes physically interconnected via the refueling canal (flooded up) with the fuel pool gates removed. Reference 18 was utilized for the distance factors used in the analyses. The analyses do not take credit for the time delay (at least 8.1 hours) between the postulated loss of cooling and the onset of boiling. Also, the analyses take no credit for hold-up or retention of radioactive airborne gases in the Reactor Building. The net effects of these two assumptions are to assume a radioactive release outside of the Reactor Building immediately upon loss of cooling.

As shown in Table 3, the worst case conservative estimate of the control room dose is only about 15% of the applicable limit. All of the other calculated doses are much lower percentages of their associated regulatory limits.

Additional analyses were performed to determine thyroid, whole body, and skin doses on the refueling level of the reactor building. These calculations assume a 70,000 scfm air change at the refueling level (corresponding to reactor building ventilation operation). The calculated thyroid dose rate could be as high as 1 Rem/hr, and could necessitate the use of breathing apparatus during the hypothetical recovery actions on the refueling level of the reactor building. However, these estimated doses are well above what would be expected were realistic assumptions to be used (Reference 19).

The proposed increase in SFP storage capacity would increase the radionuclides in the pool which would be "available" for release following a protracted loss of cooling ecvent. However, given the increase would be less than ten percent (10%), the conservative nature of the dose calculations, and the fact the predicted consequences are below 15% of the applicable regulatory limit, it is clear the increase in storage capacity would not alter the conclusions of this report from a dose consequence perspective.

9.4 DHR System Reliability Assessment

In addition to the above failure modes and effects analyses, a reliability assessment (Reference 20) was performed for the DHR system (in its nominal heat removal capability configuration) to pinpoint any potential vulnerabilities and to plan contingency measures. The results of the reliability assessment indicate that the DHR system design is highly reliable. The highest contributors to total system unreliability are associated with the concurrent failure of the pumps (either both primary side or both secondary side), or with the failure of the pressure control valve, 32PCV-100. These vulnerabilities are compensated for by provision of duplicates of the major components: primary pump, secondary pump, and heat exchanger; or manual bypass of the strainer or PCV. The selected spares are maintained onsite and therefore the time to effect the repair to restore the system to its two-loop configuration is minimized. In the event that the offsite power supply to the DHR system is not available, the DHR electrical power system would be provided by a backup truck-mounted diesel generator during refueling outages. The truck-mounted diesel generator will be hard-wired into the DHR system MCC.

9.5 Effects of Elevated Water Temperature on SFP Components

The effects of elevated SFP water temperature were evaluated as part of the previous SFP expansion (rerack). The results of these evaluations are summarized in the NRC Safety Evaluation (Reference 5) for Operating License Amendment No. 175. Those results are unaffected by the installation of the DHR system, with one positive exception: the determination of the maximum pool water temperature given a single active failure in the FPCC system.

There is no directly analogous full core offload evaluation with the DHR system installed and operating in its normal heat removal capacity (in lieu of the FPCC system), as the heatup of the SFP is mitigated through operation of the backup DHR component or train. Assuming that it takes up to two hours to effect the backup, pool heatup would be a maximum of approximately 24 ^{0}F (from an initial assumed SFP water temperature of 114 ^{0}F). Note that this maintains the water temperature below the 140 ^{0}F guideline of Standard Review Plan Section 9.1.3, reference 21. Note too, the heat load analysis performed in support of this DHR evaluation conservatively assumes that core offload is initiated 2.5 days after shutdown and completed within approximately 2.5 days (whereas the previous SFP expansion analyses assumed a 4-day delay after shutdown prior to fuel movement, and limited the transfer rate of assemblies to four per hour). Hence, it is obvious that the installation and operation of the DHR system is beneficial in limiting peak SFP water temperatures following a postulated single active failure. Since the peak SFP water temperature for the DHR case is less than the peak SFP water temperature for the FPCC case, the previous equipment evaluations are bounding and need not be re-performed for this evaluation.

10 Conclusions and Recommendations

The DHR will provide the ability to remove the decay heat from the RPV and the SFP independently of the RHR system and the FPCC system. Provision of the DHR system to augment or replace existing decay heet removal capabilities results in a significant increase in flexibility and added assurance decay heat can be removed for various transients and accidents. The installation and operation of the DHR will :

1) Provide enhanced SFP cooling capability during normal plant operation, for example, when the FPC is unavailable because of test or maintenance.

2) Provide enhanced SFP cooling capability during refueling outages.

3) Provide an independent means of removing decay heat from fuel elements in the RPV, provided the reactor cavity is flooded and the SFP gates removed.

4) Reduce the need for RHR operation in the fuel pool cooling assist mode. RHR operation in this configuration necessitates throttling of the RHR service water (RHRSW) valve on the outlet of the RHR heat exchanger in service. Throttling of the heat exchanger outlet valve is thought to be a contributing factor in the erosion/corrosion observed in the downstream RHRSW piping.

   Use of RHR in the fuel pool cooling assist mode has historically resulted in decreased clarity of SFP and reactor cavity water. Operation of the DHR prior to and during refueling would eliminate that concern.

5) Reduce the need for RHR operation in the SDC mode during refueling outages (after approximately 4.5 days).

6) Provides for additional flexibility in terms of outage planning consistent with ALARA and Shutdown Risk Management principles (Reference 22).

7) Under specified conditions, and with the appropriate administrative controls in place, both trains of RHR can safely be removed from service and the DHR utilized to remove decay heat from both the spent fuel pool and the reactor pressure vessel.

Based on the reviews summarized in this report, installation and operation of the DHR would not constitute an unreviewed safety question. Therefore, it is recommended Nuclear Safety Evaluations be developed, pursuant to 10 CFR 50.59, for the installation, testing, and operation of the DHR.

Based on operating experience with the DHR during and after RO12, it is recommended the DHR remain in service, with the attendent portable diesel generator available, subsequent to reload and until such time as the decay heat load in the SFP is within the capability of the FPCCS operating in the single pump configuration. This recommendation is considered as a prudent action and is neither a "safety requirement" of this report nor is it a requirement of the associated NSE, 96-042.

Based on evaluations performed in support of the proposed expansion in SFP storage capacity, there is no need to revise administrative controls or procedures associated with the DHR to reflect the slight (worst case) increase in SFP decay heat load. The results and conclusions of this report remain valid for the proposed increased storage capacity condition.

TABLE 1

DECAY HEAT LOADS

Approximate Days After Shutdown	Decay Heat, (10^6 BTU/HR)
1	47.9
2	39.2
3	33.6
4	29.7
5	26.9
6	24.8
7	23.2
10	20.0
20	15.1
30	12.4
40	10.6

TABLE 2

DECAY HEAT REMOVAL CAPABILITIES

Method of Decay Heat Removal	10^6 BTU/HR
DHR System (Maximum)	45
DHR System (Nominal)	30
RHR Assist + FPC (1 pump, 1 HX)	24
FPC (2 pump, 2 HXs)	10
FPC (1 pump, 2 HX)	6.3

TABLE 3

Dose Consequences Following A Postulated Loss of Decay Heat Removal

Receptor Location	Thyroid Dose,	Whole Body Rem	Skin Dose, Rem	Remarks
Site Boundary	6.2 X 10-3	5.2 X 10-5	1.5 X 10-4	
(Dose Limits:)	(300)	(25)	(N/A)	(From 10 CFR Part 100)
Low Population Zone	2.2 X 10-2	1.3 X 10-4	3.6 X 10-4	
(Dose Limits:)	(300)	(25)	(N/A)	(From 10 CFR Part 100)
Control Room	4.5	5.4 X 10-3	7.9 X 10-2	
(Dose Limits:)	(30)	(5)	(30)	(Thyroid, Skin from ≤RP 6.4; WB from 10 CFR 50 Appendix A, GDC 19)

The regulatory dose limits for a design basis Loss of Coolant Accident (LOCA) are given in parentheses. The exposure intervals assumed are two hours at the site boundary, and 30 days at the LPZ and in the Control Room. Thyroid doses are based on the dose conversion factors in TID-14844 (Reference 19).

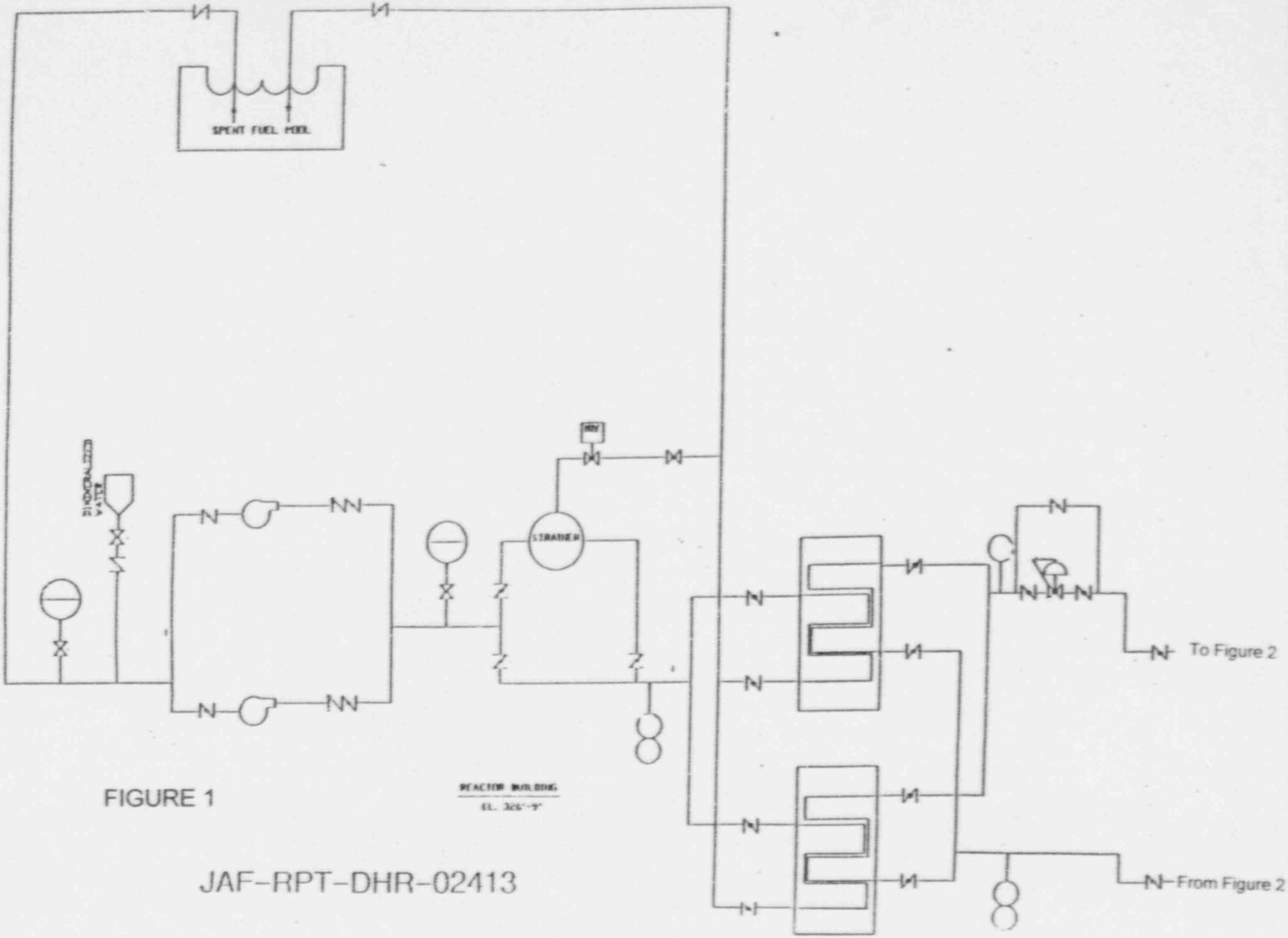


FIGURE 1

JAF-RPT-DHR-02413

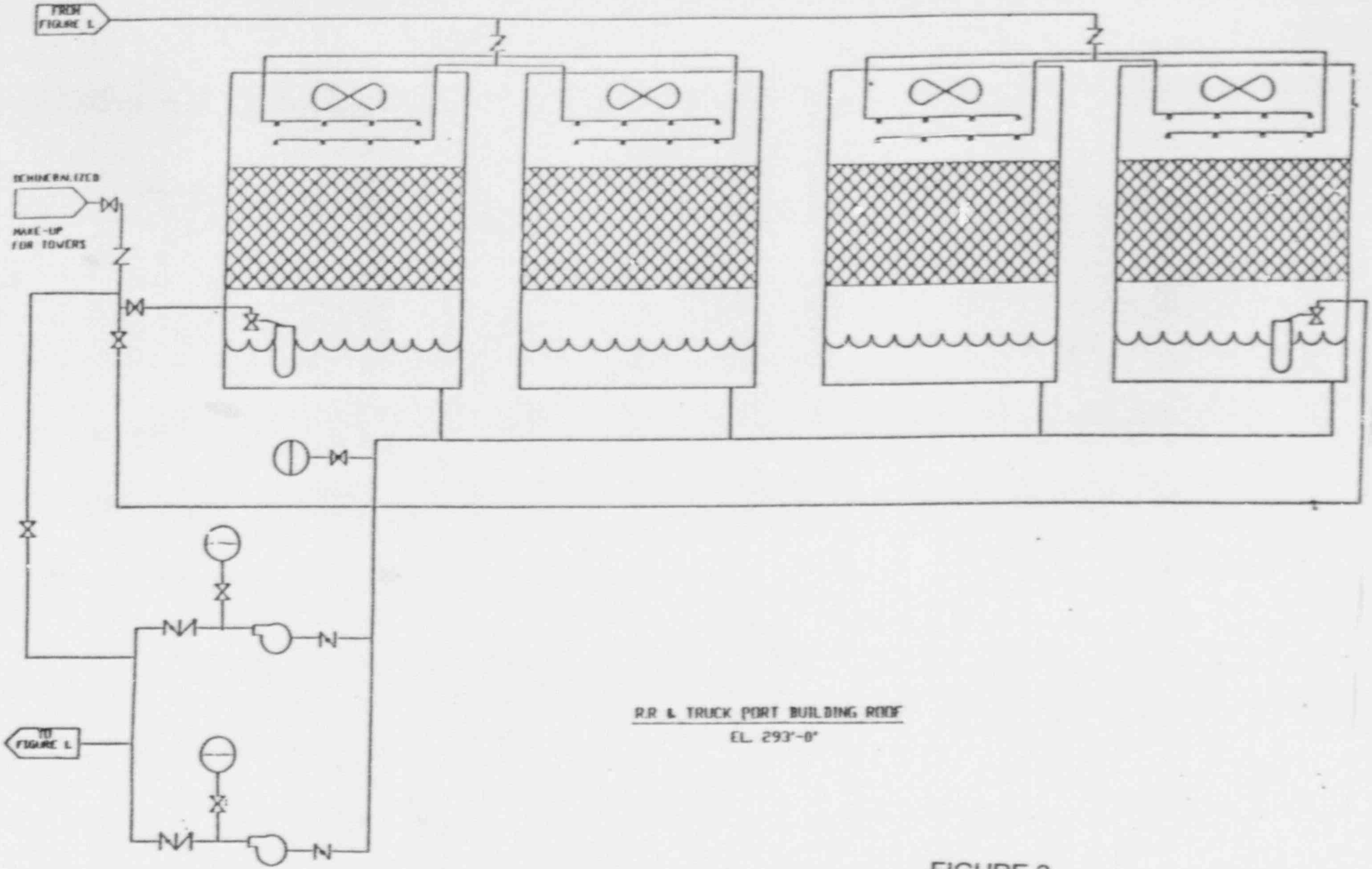


FIGURE 2

JAF-RPT-DHR-02413

FIGURE 3

JAF DHR ELECTRICAL ONE LINE

JAF-RPT-DHR-02413

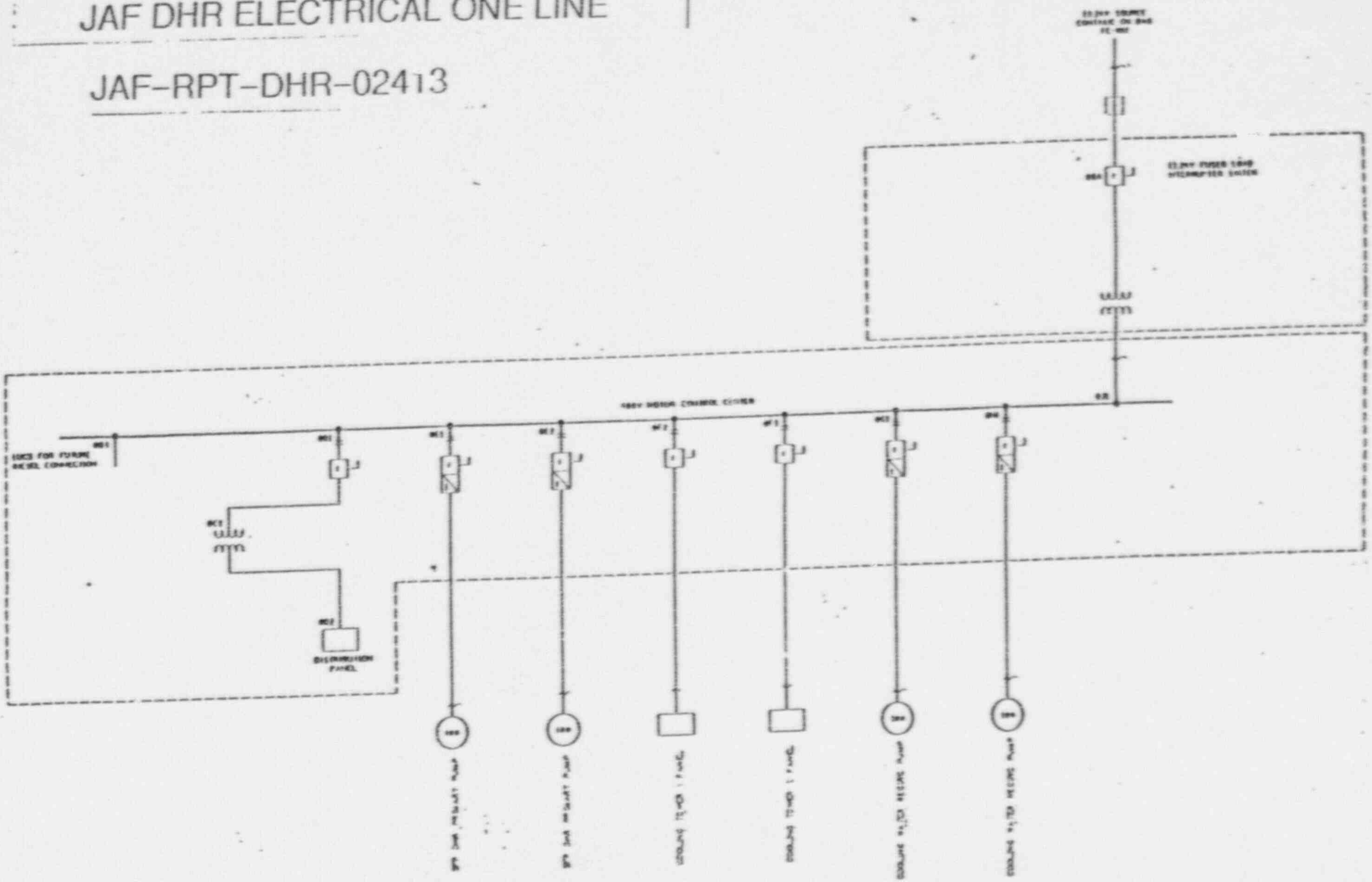
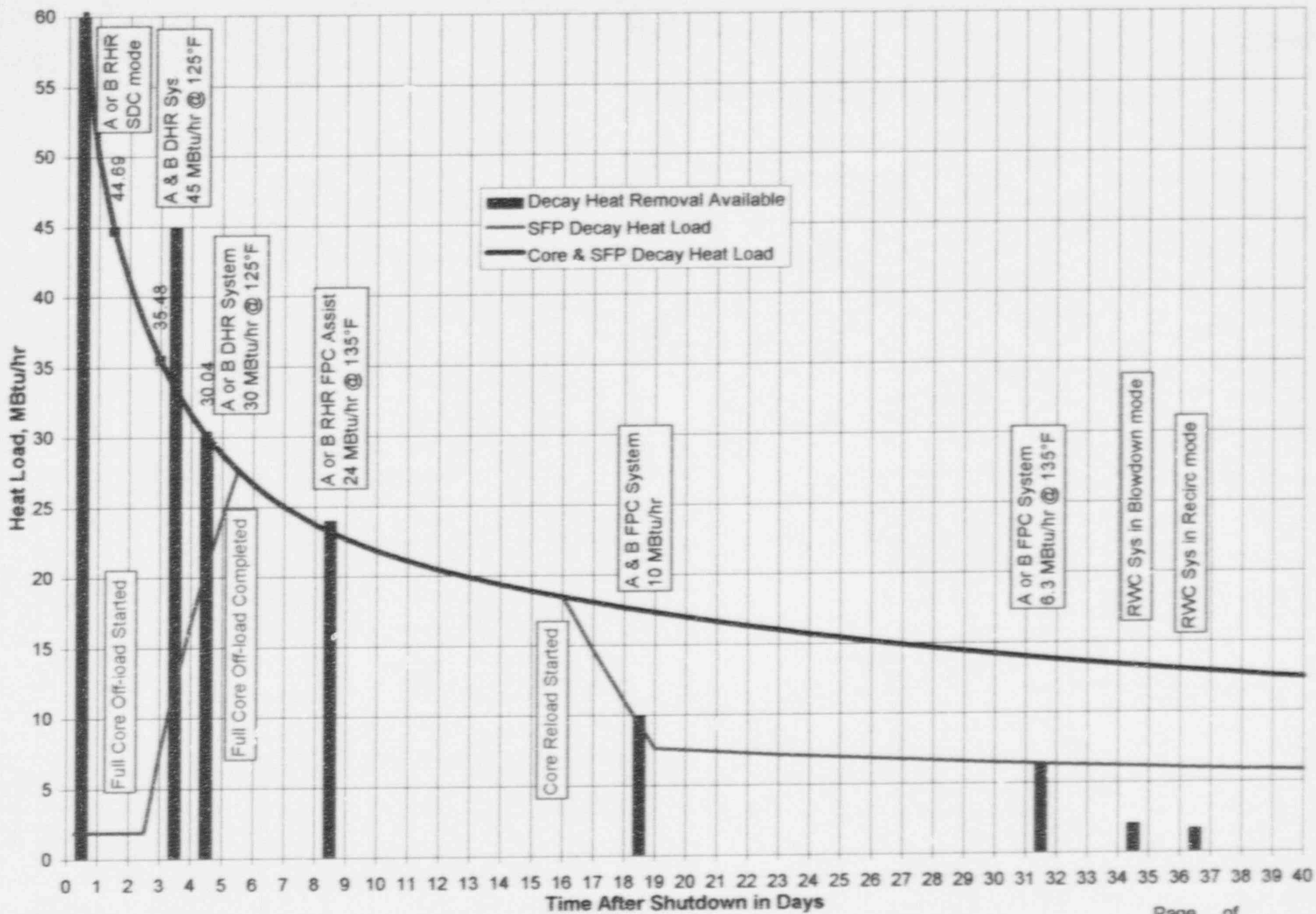
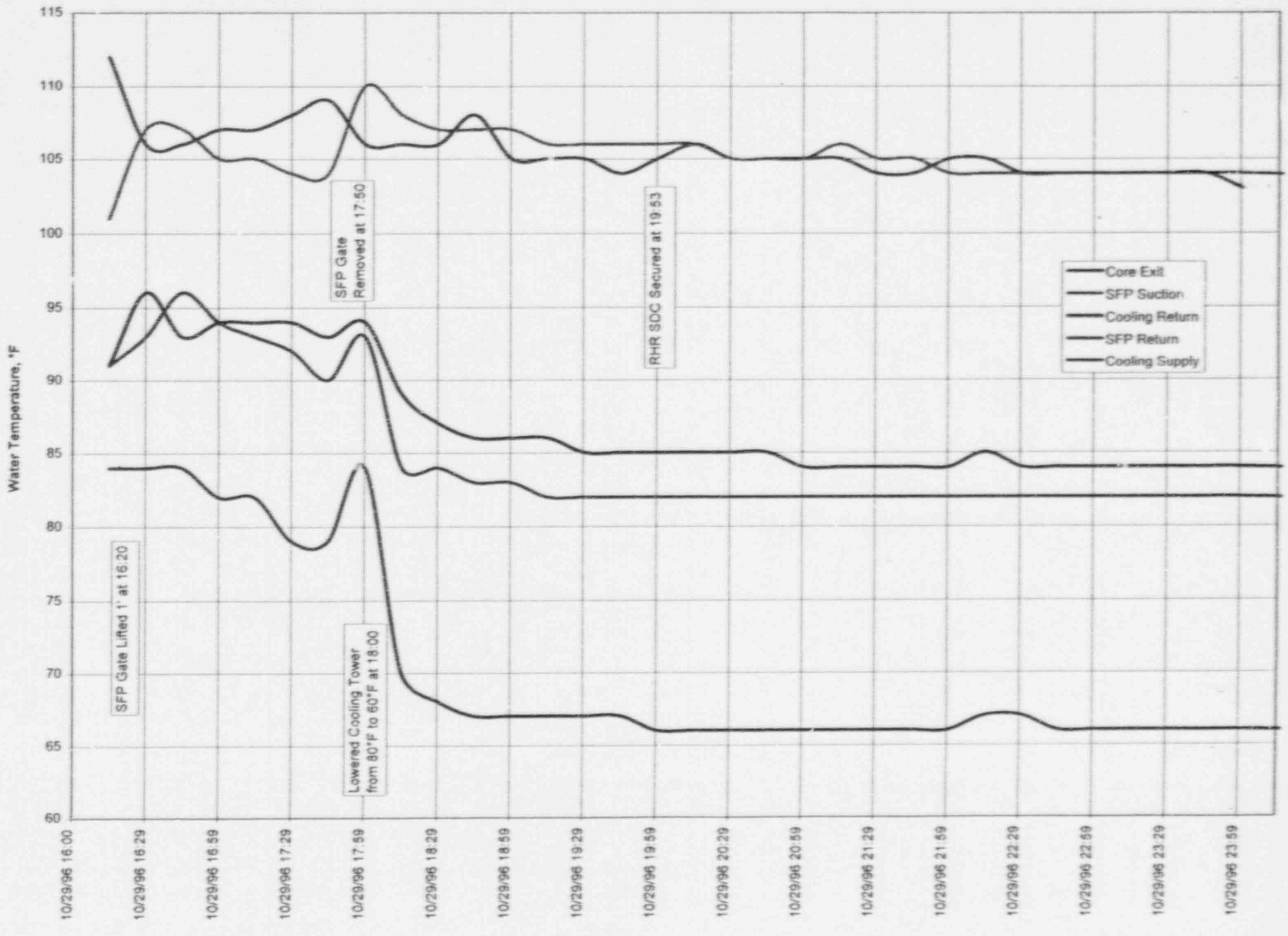


Figure 4

Decay Heat Load versus Time After Shutdown
Core and Spent Fuel Pool



POT-32C DHR System Functional Test



POT-32C DHR System Functional Test

			TI-120	TI-121A	TI-123	TI122A	FI-131	FI-132
	Time	Core Exit	SFP Suction	SFP Return	Cooling Supply	Cooling Return	SFP Flow	Cooling Flow
Cooling Tower 80°F	10/29/96 16:15	112	101	91	84	91	2450	3130
	10/29/96 16:30	106	107	93	84	96	2440	3130
	10/29/96 16:45	106	107	96	84	93	2450	3130
	10/29/96 17:00	107	105	94	82	94	2450	3130
	10/29/96 17:15	107	105	93	82	94	2450	3140
	10/29/96 17:30	108	104	92	79	94	2450	3140
	10/29/96 17:45	109	104	90	79	93	2440	3150
Cooling Tower 60°F	10/29/96 18:00	106	110	93	84	94	2440	3140
	10/29/96 18:15	106	108	84	70	89	2440	3140
	10/29/96 18:30	106	107	84	68	87	2430	3130
	10/29/96 18:45	108	107	83	67	86	2440	3120
	10/29/96 19:00	105	107	83	67	86	2440	3130
	10/29/96 19:15	105	106	82	67	86	2440	3150
	10/29/96 19:30	105	106	82	67	85	2440	3140
	10/29/96 19:45	104	106	82	67	85	2440	3150
	10/29/96 20:00	105	106	82	66	85	2410	3120
	10/29/96 20:15	106	106	82	66	85	2420	3130
	10/29/96 20:30	105	105	82	66	85	2430	3130
	10/29/96 20:45	105	105	82	66	85	2430	3150
	10/29/96 21:00	105	105	82	66	84	2420	3110
	10/29/96 21:15	105	106	82	66	84	2430	3150
	10/29/96 21:30	104	105	82	66	84	2440	3140
	10/29/96 21:45	104	105	82	66	84	2460	3150
	10/29/96 22:00	105	104	82	66	84	2420	3160
	10/29/96 22:15	105	104	82	67	85	2430	3090
	10/29/96 22:30	104	104	82	67	84	2470	3140
	10/29/96 22:45	104	104	82	66	84	2430	3140
	10/29/96 23:00	104	104	82	66	84	2470	3170
	10/29/96 23:15	104	104	82	66	84	2430	3170
	10/29/96 23:30	104	104	82	66	84	2410	3130
	10/29/96 23:45	104	104	82	66	84	2440	3140
	10/30/96 0:00	103	104	82	66	84	2450	3150
	10/31/96 14:30		92	74	62	76	2620	3130
Cooling Tower 65°F	11/1/96 8:30		94	78	66	79	2620	3110
	11/1/96 10:30		94	78	67	80	2610	3100
	11/1/96 14:45		94	80	70	80	2610	3100
	11/1/96 19:30		94	80	70	80	2610	3100
	11/2/96 14:30		91	75	64	76	2520	3130
	11/4/96 7:15		90	74	64	76	2480	3110
	11/4/96 14:00		89	73	63	75	2490	3130
	11/5/96 7:45		89	76	68	78	2500	3140
	11/5/96 17:00		90	78	72	80	2490	3140
Cooling Tower 70°F	11/6/96 8:30		93	80	73	82	2430	3110
	11/6/96 14:00		94	79	70	81	2440	3110
	11/7/96 7:30		94	80	71	81	2300	3130
	11/7/96 11:15		96	82	73	83	2280	3120
Cooling Tower 67°F	11/7/96 15:00		97	82	73	84	2260	3120
	11/7/96 16:30		97	82	74	84	2240	3130
	11/7/96 16:45		97	82	73	84	2200	3130

Attachment VII to JPN-97-033

NON-PROPRIETARY COPY OF LICENSING REPORT
FOR THE ADDITION OF STORAGE RACKS

PROPOSED TECHNICAL SPECIFICATION
CHANGES REGARDING DESIGN FEATURES

New York Power Authority

JAMES A. FITZPATRICK NUCLEAR POWER PLANT
Docket No. 50-333
DPR-59