Safety Evaluation No. 2983

RTYPE A9.03

SAFETY EVALUATION
PILGRIM NUCLEAR POWER STATION

Initiator	Group.	Department	Document No.	Calc. No.	System Name
P. Doody	NESG	S&SA		See References Section of Att. A	See Att. A

Description of Proposed change, test, or experiment: Evaluate results of a design verification for safety related cooling systems using the FSAR site maximum salt service water temperature of 75°F. Revise the FSAR to reflect the analysis results. Revise Technical Specification Bases 3/4.7.A regarding local suppression pool temperature limits (See Attachment C for markup).

SAFETY EVALUATION CONCLUSIONS:

	Yes	No	
1.	☐	☒	May the proposed activity increase the probability of occurrence of an accident previously evaluated in the Final Safety Analysis Report?
2.	☐	☒	May the proposed activity increase the consequences of an accident previously evaluated in the Final Safety Analysis Report?
3.	☐	☒	May the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report?
4.	☐	☒	May the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report?
5.	☐	☒	May the proposed activity create the possibility of an accident of a different type than any previously evaluated in the Final Safety Analysis Report?
6.	☐	☒	May the proposed activity create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the Final Safety Analysis Report?
7.	☐	☒	Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?

BASIS FOR SAFETY EVALUATION CONCLUSIONS: Please refer to Attachment A

Safety Evaluation Performed by Patrick J. Doody
Patrick Doody Date 3/25/96

Safety Evaluation No. 2983

SAFETY EVALUATION
PILGRIM NUCLEAR POWER STATION

A. APPROVAL

Comments: This safety evaluation revises the suppression pool temperature bases for TS 3.7.A. See "Analysis of Effect on Safety Functions" in Att. A.

J. Rogers 3/25/96
Discipline Division Mgr./Date

See Attached Approval Sheet for Supporting Disciplines
Supporting Discipline Division Mgr./Date

B. REVIEW/APPROVAL

☐ Comments:

J Rogers 3/25/96
S&SA Division Mgr./Date

NOTES:
1) Items (14) and (15) are not required for Safety Evaluation prepared by the Plant Department.
2) The independent technical review of Plant Department Safety Evaluations is documented in Item C below.

C. ORC REVIEW

☐ This proposed change involves an unreviewed safety question and a request for authorization of this change must be filed with the NRC prior to implementation.

☑ This proposed change does not involve an unreviewed safety question.

ORC Chairperson J A Leery Date 3/29/96

ORC Meeting Number 96-16

cc:

Safety Evaluation No. 2983

Supporting Discipline Department Mgr Approvals

Thomas White, 3/25/96
Mechanical/Civil/Structural Department
Mgr./Date

Edward T Almeida 3/25/96
Instrumentation & Control Department
Mgr./Date

Swapan Das for BC 3/25/96
Electrical Engineering Department
Mgr./Date

D. FSAR Review Sheet

List FSAR text, diagrams, and indices affected by this change and corresponding FSAR revision.

Affected FSAR Section	Preliminary revision to the affected FSAR Section is shown on:
1.6	~~Attachment 1~~
4.8	~~Attachment 2~~
5.2	~~Attachment 3~~
10.5	~~Attachment 4~~
14.5	~~Attachment 5~~
14.7	~~Attachment 6~~

NOTE

See Attachment B for complete FSAR markup, new figures, tables and text.

PRELIMINARY FSAR REVISION (to be completed at time of Safety Evaluation preparation).

Prepared by: Patrick J. Doody Patrick Doody Date: 3/25/96

Approved by: [signature] Date: 3/25/96

FINAL FSAR REVISION - Prepared in accordance with NOP83E4 following operational turnover of related systems, structures, or components for use at PNPS.

E. SAFETY EVALUATION WORK SHEET

A. System/Component Failure and Consequence Analyses.

System/Component	Failure Modes	Effects of Failure	Comments
See Attached Table for Failure Modes and Consequence Analysis.			

General Reference Material Review

FSAR SECTION	PNPS TECH. SPECS	CALCULATIONS DESIGN SPECS. PROC.	REGULATORY GUIDES STANDARDS CODES
1.5 to 1.6	3.5	See "References"	See "References"
4 to 8	3.7	Section I of Attachment A	Section I of Attachment A
10			
14			
App R			

B. For the proposed hardware change, identify the failure modes that are likely for the components consistent with FSAR assumptions. For each failure mode, show the consequences to the system, structures, or related components. Especially show how the failure(s) affects the assigned safety basis (FSAR text for each system) or plant safety functions (FSAR Chapter 14 and Appendix G.)

Prepared by Patrick J. Doody Patrick Doody Date 3/25/96

System/Component Failure and Consequence Analyses.

	System or Component	Failure Modes	Effects of Failure	Comments
1.	Integrated Core and Containment Cooling	Battery failure	Leaves only one Core Spray pump, two LPCI pumps, and ADS to mitigate the LOCA.	This is the limiting failure with respect to PCT for all but the largest recirculation line breaks per the core cooling analysis. The systems available after this failure provide an acceptable CSCS response. With respect to containment cooling, this failure leaves a couple of satisfactory options. Using EOPs the containment cooling option utilized will depend on vessel level indication during the response (i.e., 2/3 core height < level indication <TAF, or level indication > TAF). If level can be raised above TAF and maintained with a single Core Spray, the RHR pumps can be taken out of LPCI mode and used in SPC mode or containment spray. A response such as this is assumed to be used in the steam line break analysis. If level cannot be raised and maintained above TAF, the break flow out the vessel is substantial. LPCI with heat rejection can be utilized for liquid breaks inside primary containment of sufficient size to support continuous recirculation (including the design basis LOCA) as described in Attachment A. In the case of small liquid breaks, where level can be maintained with a single Core Spray, the RHR pumps can be taken out of LPCI mode and used in SPC mode or containment spray.
2.	Integrated Core and Containment Cooling	LPCI injection valve on unbroken recirculation loop fails to open	Leaves two Core Spray, HPCI, and ADS. LPCI and LPCI with heat rejection unavailable	This failure is the limiting failure for the recirculation line break with respect to PCT per the core cooling analysis. The systems available after this failure provide an acceptable CSCS response. This failure is less limiting than a battery failure for containment cooling because both loops of RHR suppression pool cooling and containment spray are available. Since, two RHR heat exchangers with the associated support systems can provide a much lower temperature response for containment, component cooling, and compartment cooling, this is clearly not the limiting failure for containment cooling.
3.	Integrated Core and Containment Cooling	Loss of one emergency diesel generator	Leaves one Core Spray pump, two LPCI pumps, and ADS for LOCA mitigation.	The effects of this failure are the same as item 1 in this table.
4.	Integrated Core and Containment Cooling	HPCI failure	Leaves all other ECCS and Containment Cooling systems.	This is a significant but acceptable failure for small line breaks that do not rapidly depressurize the vessel per the core cooling analysis. The ADS system is relied on to depressurize the reactor vessel so the low pressure systems (Core Spray or LPCI) can be used to cool the core and maintain long-term cooling. For containment cooling purposes, this failure is less severe than the battery failure because two loops of containment cooling are available. Please refer to the comments for item 1 and 2 in this table.

Attachment A
Safety Evaluation No. 2983

Figures

A. Purpose

Safety-related systems, structures and components (SSC's) depend upon cooling systems to maintain temperatures in the plant at acceptable levels during design basis events.

The current FSAR analysis for design basis events and the performance of emergency cooling equipment uses a constant ultimate heat sink temperature of 65°F. A design verification of the plant safety analysis design basis was performed using a constant SSW injection temperature of 75°F as documented in references 20, 22, 34, 35, 37, 39 to 45, 54, and 55. This document evaluates the results of the design verification against 10CFR50.59 criteria.

Also, this evaluation documents proposed changes to the FSAR which incorporate results from the design verification consistent with the scope of information and level of detail in the current FSAR. The new analysis presented in this evaluation and the accompanying FSAR changes will supplement rather than replace the current analysis.

B. Description of Change(s)

The design value for the SSW injection .emperature directly or indirectly affects information presented in the following FSAR sections:

FSAR Section 4.8	Residual Heat Removal System
FSAR Section 5.2	Primary Containment System
FSAR Section 6.4	Core Standby Cooling System
FSAR Section 10.5	Reactor Building Closed Cooling Water System
FSAR Section 10.7	Salt Service Water System
FSAR Section 10.18	Equipment Area Cooling System
FSAR Section 14.5	Accident Analysis

PNPS has been evaluated and in some cases modified pursuant to rule making for Fire events (10 CFR 50 Appendix R), Environmental Qualification (10CFR50.49) and ATWS events (10 CFR 50.62) and Station Blackout Rule (10 CFR 50.63). These events were included in the scope of the design verification to the extent that the analysis previously performed for these events is affected by the design value of the SSW injection temperature.

C. Systems, Structures, Components Affected

The following safety systems and associated components are affected:

1. Residual Heat Removal (RHR) System
2. Primary Containment System
3. Core Spray (CS) System
4. Salt Service Water (SSW) System
5. Reactor Building Closed Cooling Water (RBCCW) System
6. Equipment Area Cooling (EAC) System

D. Systems, Structures and Components Indirectly Affected

1. Standby Gas Treatment System
2. Auxiliary Power Distribution System
3. Standby AC Power Source
4. Instrumentation and Control System(s)

E. Safety Functions of Affected Components

The temperature of the ultimate heat sink is a significant determining factor for the heat removal rate achieved by the cooling train(s) during shutdowns or the design basis events. For the purpose of this evaluation, safety-related cooling is broken down into four basic functions:

Core Cooling
Containment Cooling
Component Cooling
Building Compartment Cooling

Although these different functions are often analyzed independently, they are mutually dependent as discussed below. The reliability and/or performance of systems which perform each of the basic cooling functions is directly or indirectly dependent on the SSW inlet temperature.

As illustrated in Figure 1 the emergency cooling functions utilize four different basic methods:

Air Exchange (ventilation using outside air)
Direct Cooling (submergence or spray directly on the component)
Air Cooling (air to water heat exchange)
Water Cooling (water to water or oil to water heat exchange)

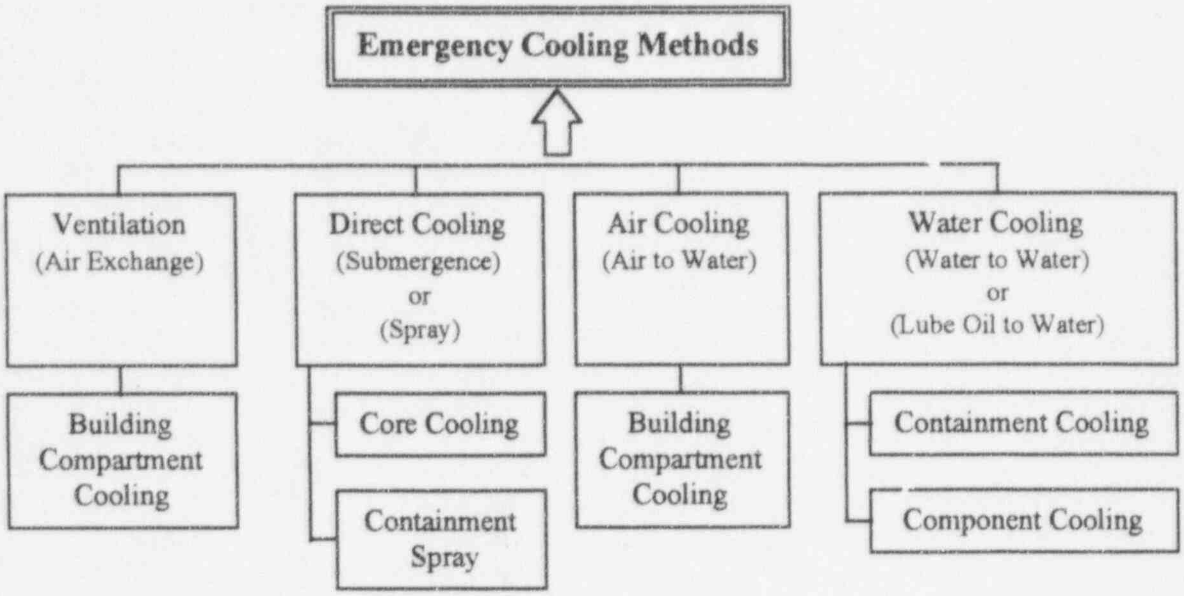

Figure 1

The Four Basic Cooling Functions

Core cooling is essential to protect the integrity of the fuel clad, minimize metal-water reaction, and ensure a coolable geometry. The fuel clad is the first of four essential barriers protecting against the release of radioactive material. Figure 2 presents a three tiered diagram that contains core cooling performance criteria on the first level, systems that perform the function on the second level, and supporting safety functions of containment, component, and compartment cooling on the third level. As illustrated on Figure 2, the following systems and support functions are relied on to perform the core cooling safety function:

a) RHR - Low Pressure Coolant Injection Mode[1]

b) Core Spray[1]

c) High Pressure Coolant Injection

The support functions are:

d) Containment Cooling

e) Component Cooling

f) Compartment Cooling

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[1] Core Spray and LPCI perform the core cooling function in conjunction with the Alternate Depressurization System (ADS) during loss of coolant accidents that don't depressurize the reactor vessel rapidly enough to ensure adequate core cooling.

Containment cooling is essential to preserve containment integrity by maintaining the temperature and pressure of the containment structure below the design limits. After an unisolable failure of the reactor coolant pressure boundary inside containment (PBIC), the containment is the second barrier protecting against the release of radioactive material. Figure 3 presents a three tiered diagram that contains containment cooling performance criteria on the first level, systems that perform the function on the second level, and supporting safety functions of component, and compartment cooling on the third level. As illustrated on Figure 3, the following systems and support functions are relied on to perform the containment cooling safety function:

a) RHR - Suppression Pool Cooling
LPCI with Heat Rejection
Containment Spray

The support functions are:

a) Component Cooling

b) Building Compartment Cooling

Component cooling refers to the cooling provided to active and passive components by the Reactor Building Closed Cooling Water (RBCCW) system. Component cooling does not directly cool any of the radioactive material barriers, however it is essential to assure reliable operation of the active equipment performing both core and containment cooling and to remove heat from the containment via the RHR and RBCCW heat exchangers. Figure 4 diagrams component cooling criteria and the primary systems that are relied on to perform the component cooling function.

Building compartment cooling includes the cooling provided by the Equipment Area Coolers located in various compartments of the plant. These coolers include fans which circulate the compartment atmosphere across cooling coils that contain water circulating in the Reactor Building Closed Cooling Water (RBCCW) system. Building compartment cooling also includes the ventilated motor control center (MCC) enclosures B17, B18, B20. The temperature inside these MCC enclosures is maintained by ventilation fans that circulate air from outside the enclosure (i.e., from general area of floor elevation 23 foot of the Reactor Building) through the enclosure.

Building compartment cooling is essential to maintain the building compartment ambient temperature(s) within acceptable limits by removing heat transmitted to the building air from fluid system piping, the containment structure, and electrical equipment. Figure 5 diagrams compartment cooling criteria and the systems relied on to perform the building compartment cooling function.

Core Cooling

Overall Performance Criteria
Peak Clad Temperature < 2200°F
Core Wide Metal-Water Reaction < 0.1 %
Maximum Clad Oxidation < 0.17 %
Coolable Geometry
Long-Term Cooling Maintained

Core Standby Cooling Systems

Core Spray*
CS

Low Pressure Coolant Injection*
LPCI

High Pressure Coolant Injection
HPCI

* LPCI and CS pumps act in conjunction with ADS for breaks that do not depressurize the reactor vessel

Core Standby Cooling Support Functions

Containment Cooling
Criteria
Peak Liner Temperature < 281°F
Peak Pressure < 56 psig
NPSHA > NPSHR for ECCS Pumps [Limiting]

Equipment Requirements
Associated Pumps
Associated Heat Exchangers
Associated Valves
Associated Support Systems [See Fig 3]

Component Cooling
Criteria
Cooling Water Inlet Temp < Maximum Allowed
Cooling Water Inlet Flow Rate > Minimum Required

Equipment Requirements
Associated Heat Exchangers and Coolers
Associated Support Systems [See Fig 4]

Compartment Cooling
Criteria
Peak Compartment Temperature and
Integrated Exposure to Elevated Temperature
Less Than
Component EQ Test Data

Equipment Requirements
Associated Coolers or Ventilation Systems
Associated Support Systems [See Fig 5]

Figure 2

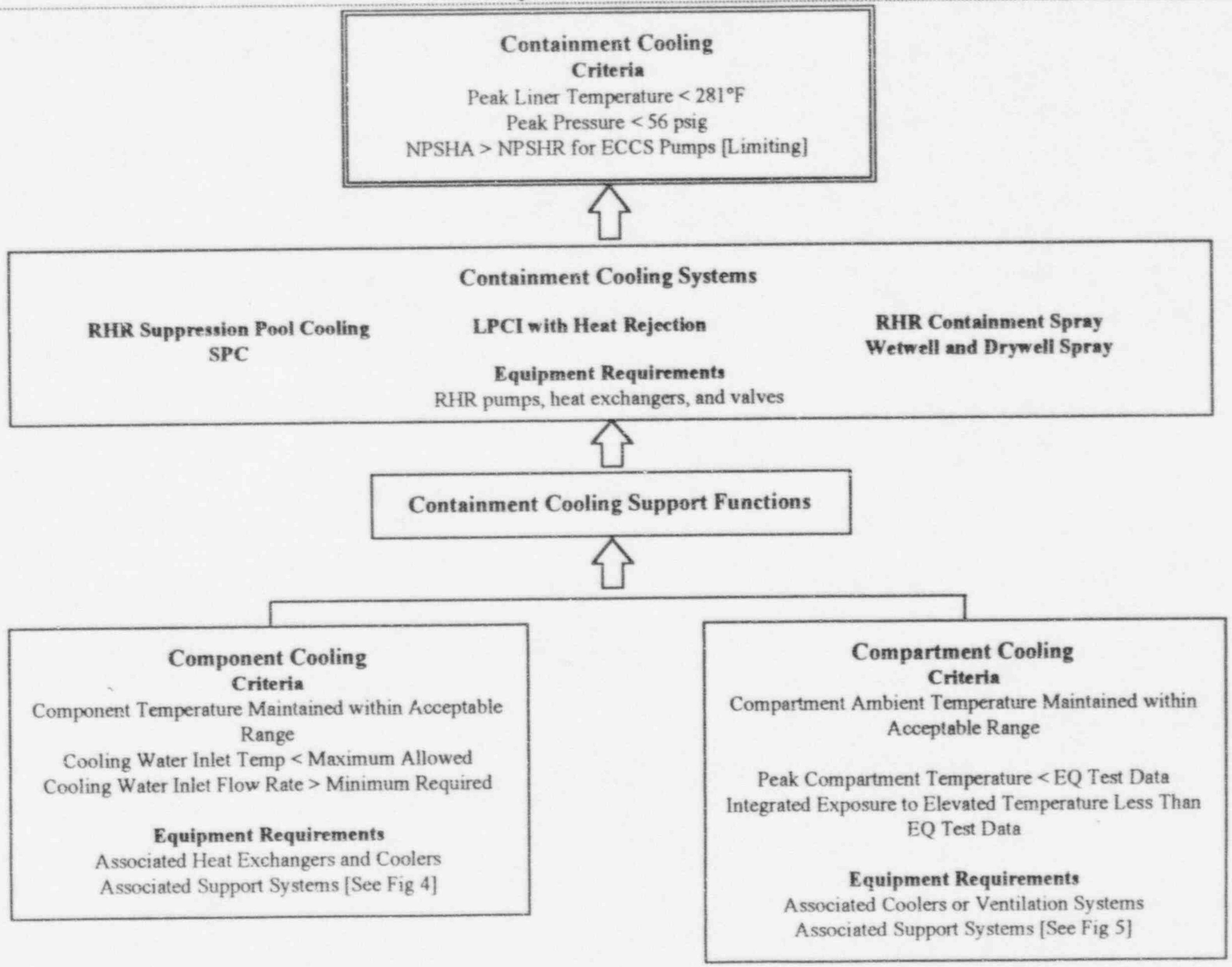
Containment Cooling
Criteria
Peak Liner Temperature < 281°F
Peak Pressure < 56 psig
NPSHA > NPSHR for ECCS Pumps [Limiting]
Containment Cooling Systems
RHR Suppression Pool Cooling
SPC
LPCI with Heat Rejection
RHR Containment Spray
Wetwell and Drywell Spray
Equipment Requirements
RHR pumps, heat exchangers, and valves
Containment Cooling Support Functions
Component Cooling
Criteria
Component Temperature Maintained within Acceptable Range
Cooling Water Inlet Temp < Maximum Allowed
Cooling Water Inlet Flow Rate > Minimum Required
Equipment Requirements
Associated Heat Exchangers and Coolers
Associated Support Systems [See Fig 4]
Compartment Cooling
Criteria
Compartment Ambient Temperature Maintained within Acceptable Range
Peak Compartment Temperature < EQ Test Data
Integrated Exposure to Elevated Temperature Less Than EQ Test Data
Equipment Requirements
Associated Coolers or Ventilation Systems
Associated Support Systems [See Fig 5]

Figure 3

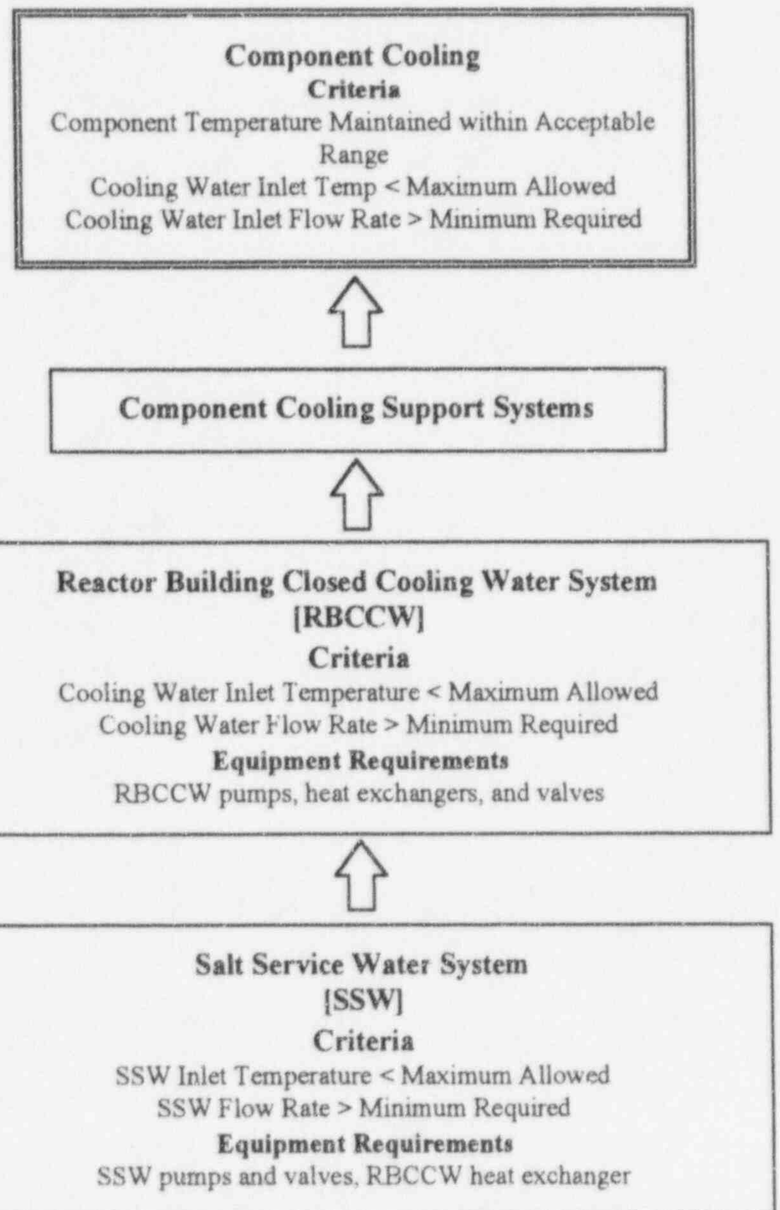
Component Cooling
Criteria
Component Temperature Maintained within Acceptable Range
Cooling Water Inlet Temp < Maximum Allowed
Cooling Water Inlet Flow Rate > Minimum Required
Component Cooling Support Systems
Reactor Building Closed Cooling Water System
[RBCCW]
Criteria
Cooling Water Inlet Temperature < Maximum Allowed
Cooling Water Flow Rate > Minimum Required
Equipment Requirements
RBCCW pumps, heat exchangers, and valves
Salt Service Water System
[SSW]
Criteria
SSW Inlet Temperature < Maximum Allowed
SSW Flow Rate > Minimum Required
Equipment Requirements
SSW pumps and valves, RBCCW heat exchanger

Figure 4

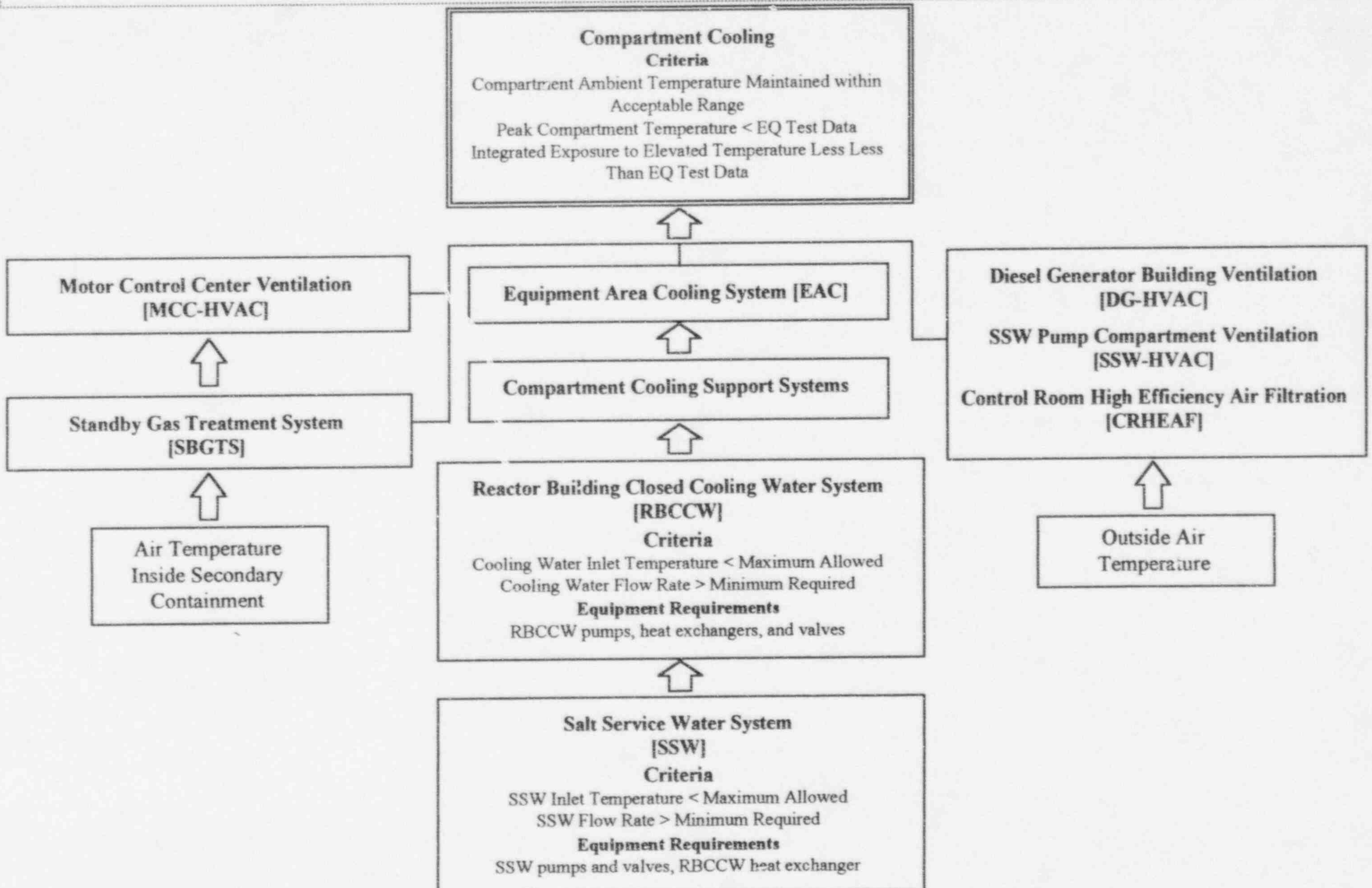
Compartment Cooling
Criteria
Compartment Ambient Temperature Maintained within Acceptable Range
Peak Compartment Temperature < EQ Test Data
Integrated Exposure to Elevated Temperature Less Less Than EQ Test Data
Motor Control Center Ventilation [MCC-HVAC]
Equipment Area Cooling System [EAC]
Diesel Generator Building Ventilation [DG-HVAC]
SSW Pump Compartment Ventilation [SSW-HVAC]
Control Room High Efficiency Air Filtration [CRHEAF]
Compartment Cooling Support Systems
Standby Gas Treatment System [SBGTS]
Reactor Building Closed Cooling Water System [RBCCW]
Criteria
Cooling Water Inlet Temperature < Maximum Allowed
Cooling Water Flow Rate > Minimum Required
Equipment Requirements
RBCCW pumps, heat exchangers, and valves
Air Temperature Inside Secondary Containment
Outside Air Temperature
Salt Service Water System [SSW]
Criteria
SSW Inlet Temperature < Maximum Allowed
SSW Flow Rate > Minimum Required
Equipment Requirements
SSW pumps and valves, RBCCW heat exchanger

Figure 5

The following discussion provides the FSAR requirements for systems that perform cooling for the core, containment, components, and building compartments.

1. **Residual Heat Removal (RHR) System [Ref. 2]**

The principal safety function(s) of the RHR system are emergency core cooling and decay heat removal. The RHR system has four basic modes of operation:

(a) Low Pressure Coolant Injection (LPCI) - provides makeup water to the reactor vessel after a loss-of-coolant-accident (LOCA) to cool the fuel clad and limit the clad damage due to overheating.

(b) Suppression Pool Cooling (SPC) - provides heat removal from the primary containment system via the suppression pool to maintain the long term containment pressure and temperature below the design limits.

The suppression pool cooling subsystem is placed in operation to limit the temperature of the water in the suppression pool following a LOCA. In SPC mode the RHR pumps draw water from the suppression pool and pump the water through the RHRS heat exchanger where cooling takes place by transferring heat to the RBCCW system. The fluid is then discharged back to the suppression pool.

The LPCI with heat rejection mode is a form of suppression pool cooling used in EOPs where the RHR system performs the LPCI and SPC functions simultaneously. Long-term core and suppression pool cooling can be performed for intermediate to large liquid breaks inside primary containment including the design basis LOCA by use of the LPCI with Heat Rejection mode. The RHR pumps draw water from the suppression pool and pump the water through the RHR heat exchanger where cooling takes place by transferring heat to the RBCCW system. The fluid is then discharged back to the reactor vessel where heat from the reactor core and vessel is transferred to the fluid before it flows out of the pipe break and returns to the suppression pool via the vent system which connects the drywell to the torus.

(c) Containment Spray - provides direct cooling of the containment airspace by spray of suppression pool water (cooled by the RHR heat exchanger) into the drywell airspace or wetwell airspace. This capability is in excess of the required energy removal capability for a DBA-LOCA since a super heated steam environment in the drywell does not result from this event and the containment shell design limit is not challenged.

This mode is essential for small steamline breaks which will cause a super heated steam environment in the drywell. Drywell spray absorbs the super heat energy and prevents the containment shell in the drywell from exceeding its design limit of 281°F.

(d) Shutdown Cooling (SDC) - shutdown cooling provides a means of reaching cold shutdown rapidly during a shutdown evolution by closed loop circulation of reactor inventory through the RHR heat exchanger. The SDC mode of the RHR system is not relied on to provide an active safety function in plant safety analysis. However, the SDC subsystem is capable of reducing reactor water temperatures to 125°F approximately 20 hours after reactor shutdown so that the reactor can be refueled or serviced.

2. Primary Containment System [Ref. 3]

The primary containment provides:

(a) a pressure vessel and suppression pool to contain the mass and energy released from the reactor pressure vessel (RPV) following a LOCA.

(b) a pressure vessel and suppression pool to contain noncondensible gas and energy resulting from metal water-reactions and other chemical reactions that can accompany a LOCA.

(c) a barrier against the release of radioactive material to the secondary containment and environs.

The suppression pool provides:

(a) a source of water for emergency core cooling function provided by HPCI, Core Spray and RHR-LPCI mode.

(b) a body of water used to store heat energy until it can be transferred to the ultimate heat sink by the containment cooling train(s).

3. Core Spray System [Ref. 5]

Provides core cooling by spraying water onto the core to meet core cooling criteria after a loss-of-coolant-accident (LOCA).

4. Reactor Building Closed Cooling Water System [Ref. 9]

The RBCCW system:

Provides a continuous supply of cooling water to the CSCS, the EAC system, and the RHR heat exchangers under all accident and transient conditions.

a) Provides a continuous supply of cooling water to heat exchangers, equipment coolers, and equipment area coolers to

facilitate the transfer of heat from these sources to the ultimate heat sink via the RBCCW heat exchangers.

The RBCCW system maintains containment, equipment, and compartment temperature(s) within acceptable limits by removing heat from the following :

RHR heat exchanger(s)

RHR Pump Seal Cooler(s)

Core Spray Motor Cooler(s)

Equipment Area Coolers

5. **Salt Service Water System [Ref. 10]**

The Salt Service Water system provides a continuous supply of cooling water to the secondary side of the RBCCW heat exchangers adequate for the requirements of the RBCCW under normal shutdown, transient, and accident conditions. Cooling water is taken from the intake structure by the five service water pumps and discharged into the seal well.

6. **Equipment Area Cooling System [Ref. 11]**

The Equipment Area Cooling system maintains local building compartment temperatures within acceptable limits to assure continuous reliable operation of safety related CSCS electrical components and provides cooling of building compartments by transferring heat energy from the compartment air to the reactor building closed cooling water lcop and circulates cooler air in the compartment.

7. **Standby Gas Treatment System [Ref. 4]**

The Standby Gas Treatment System is part of the secondary containment system and in conjunction with the primary containment and other emergency safeguards acts to minimize the release of radioactive material from an accident so that the offsite dose will be below the guideline values stated in 10CFR100. The SBGTS maintains the secondary containment at a negative pressure relative to the environs to prevent a ground level release of radioactive material. The SBGTS takes a suction from the reactor building, drawing the effluent through a high efficiency particulate absorber (HEPA), two charcoal beds and a final HEPA filter and discharges the effluent to the environs via an elevated release point.

8. Standby AC Power Source [Ref. 8]

The standby AC power system is described in FSAR Section 8.5 and includes two independent diesel generator sets which provide a single failure proof source of onsite power capable of providing adequate power to meet all core, containment, component, and building compartment cooling needs to safely shutdown the reactor following all transients and accidents.

Each standby diesel generator is capable of providing sufficient power to its emergency bus to satisfy the essential loads.

Each diesel generator set has the load carrying ability described in Table 8.5-3. The expected accident loads are tabulated for each diesel generator set in Table 8.5-1. The expected accident loads tabulated in Table 8.5-1 are less than the diesel generator capability provided in Table 8.5-3.

9. Auxiliary Power Distribution System [Ref. 8]

Auxiliary power distribution includes the system of electrical equipment that distributes electrical power for the operation of active safety related equipment including pumps, valves, control logic, instrumentation etc.

10. Instrumentation [Ref. 6]

Instrumentation includes safety related instrumentation used to detect abnormal plant conditions, initiate protective action, and provide indication of plant conditions during and following design basis events.

Design Basis Events

There are three general categories for internal events: transient, accident, and other events. The systems identified above are required to perform the four basic cooling functions and maintain temperature(s) within acceptable limits during each of the design basis events. Figure 6 diagrams the most limiting events (bounding) with respect to cooling requirements in each category.

The safety analysis performed for each bounding design basis event is based on the design capability of the above safety systems and is intended to demonstrate that:

1. the primary barriers against the release of radioactive material will remain intact (i.e., fuel clad, reactor coolant pressure boundary, primary containment, and secondary containment).
2. Each safety system will reliably perform its intended safety function either in the direct protection of a primary barrier or in a support function.

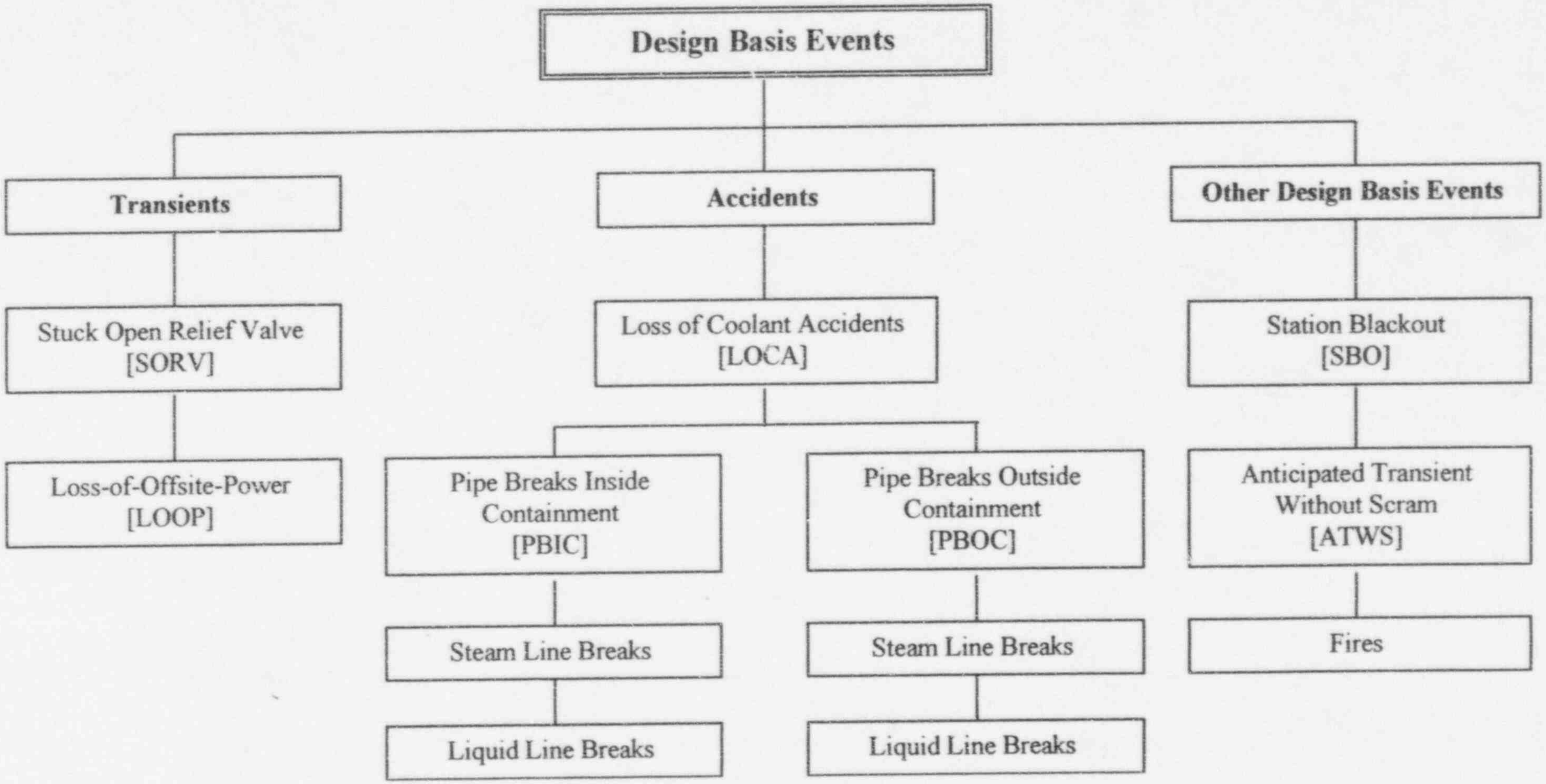
Design Basis Events
Transients
Accidents
Other Design Basis Events
Stuck Open Relief Valve [SORV]
Loss-of-Offsite-Power [LOOP]
Loss of Coolant Accidents [LOCA]
Pipe Breaks Inside Containment [PBIC]
Pipe Breaks Outside Containment [PBOC]
Steam Line Breaks
Liquid Line Breaks
Steam Line Breaks
Liquid Line Breaks
Station Blackout [SBO]
Anticipated Transient Without Scram [ATWS]
Fires

Figure 6

F. Effect on Safety Function(s)

The primary effect of an increased SSW injection temperature is an increase in the closed cooling loop water temperature and corresponding temperature increase in the suppression pool, and other areas of the plant influenced by the closed cooling water loop temperature. The location and effect(s) of these temperature increases are as follows:

1. Primary Containment:

   Increased primary containment airspace temperature and pressure for a period of time during the accident or transient response.

   Increased bulk suppression pool temperature for a period of time during the accident or transient response.

   Increased containment spray temperature and a corresponding reduction in cooling capability of the spray flow.

   Slight decrease in metal-water reaction capability of the containment and containment cooling systems.

2. Residual Heat Removal System:

   Increased temperature at inlet and outlet of RHR heat exchanger(s) for a period of time during the transient or accident response.

   Higher duty on the RHR heat exchanger for a period of time during the accident or transient response.

   Increased containment spray temperature and a corresponding reduction in cooling capability of the spray flow.

   Greater amount of time is needed to reach cold shutdown using the SDC mode of the RHR system.

3. Core Standby Cooling System(s):

   Higher temperature water delivered to the reactor for core cooling over a period of time during the accident or transient response.

   Reduced net positive suction head available at the point of minimum margin during the accident or transient response.

4. RBCCW System:

   Increased temperature of cooling water at heat exchangers and various equipment inlet(s) and outlet(s) in the RBCCW cooling loop over a period of time during the accident or transient response.

   Higher duty on the RBCCW heat exchanger over a period of time during the accident or transient response.

Higher water inlet temperatures to the component coolers and higher air and water inlet temperatures to the equipment area coolers over a period of time during the accident or transient response.

5. Salt Service Water System:

   Increased temperature of cooling water at inlet and outlet of SSW cooling loop over a period of time during the accident or transient response.

6. Equipment Area Cooling:

   Increased building compartment temperatures in those compartments that receive active cooling, as well as those that do not, over a period of time during the accident or transient response.

7. Standby Gas Treatment System.

   The air temperature in building compartments from which the SBGTS takes suction must remain below the temperature limit that causes the SBGTS train(s) to automatically shut down.

8. Standby AC Power Source

   Assumptions from the Chapter 14 design basis LOCA analysis for PNPS at 65°F SSW inlet temperature are reflected in the diesel loading (FSAR Table 8.5-1) as follows:

   - 10 minutes after the accident one RHR pump is shutoff and one additional SSW and RBCCW pump are started to support containment cooling initiation.

   The analysis performed for the design basis LOCA using a 75°F SSW injection temperature assumed a different containment cooling method than was assumed in the above analysis which combines core and containment cooling in a mode designated LPCI with Heat Rejection.

   Use of the LPCI with Heat Rejection mode results in a increased diesel generator load for a period of time during the accident response.

9. Power and Instrumentation Systems

   Increased compartment or enclosure temperature over a period of time during the accident response.

G. Analysis of Effect on Safety Functions

1. Design Value Selection

The ultimate heat sink temperature selected for the design rating of safety related heat removal systems and equipment is 65°F. This value corresponds to the highest average monthly mean heat sink temperature which historically occurs in August. FSAR Figure 2.4-2 charts the average monthly mean temperatures for each month of the year, and indicates that the average August mean is 65°F. This figure also indicates a maximum temperature of 75°F occurring in August.

Analysis of the design basis LOCA typically continues past the peak suppression pool temperature which occurs during the first six hours to include the entire 30 day cooldown process. For evaluations focused on long-term response, an average heat sink temperature of like duration is warranted. The selection of 65°F as the design value was most likely based on this rationale.

The design verification was performed using a constant 75°F temperature over the entire 30 day design basis accident response. The following presentation of site specific temperature data demonstrates that 75°F represents an upper-bound analytical value rather than an expected maximum over the course of any design basis event.

The ultimate heat sink temperature selected for the design rating of the main heat sink (e.g. main condenser) and associated equipment required for power generation is 75°F. The selected design value for power generation ensures the capability to operate without power reductions or inadvertent trips during the warmer summer months.

An in-depth analysis of the ultimate heat sink temperature was performed as part of this design verification [Ref. 33]. The purpose of this analysis was to collect and analyze historical sea water temperature data for the PNPS site and to evaluate the confidence level that the 75°F value used in this design verification will not be exceeded over a specified duration.

Seventeen years of historical sea water temperatures from 1976 to 1995, including more than 30,000 hourly measurements were analyzed. Using 17 years of data, and based on the average temperature over one hour, the mean of the maximum average temperature for a range of durations from 1 to 720 hours was calculated. The data was also statistically analyzed for scatter around the mean for the same range of durations. Adjustments were made to account for measurement error and data gaps. Assuming the sample data parameters approximate their corresponding parameters for a normally distributed population, the proposed site maximum ultimate heat sink temperature of 75°F will not be exceeded in any given year with a confidence level of approximately 95%. This confidence level means temperature

measurements at PNPS in excess of 75°F are expected to occur in one year out of twenty. In that year, the temperature may exceed 75°F on more than one occasion but not for extended periods of time.

This analysis of site specific temperature measurements at the plant intake indicates a high confidence that the selected value for the maximum site SSW inlet temperature will not be exceeded for an extended period of time. It is important to note that this analysis does indicate that 75°F could be exceeded on occasion, however the duration of any period of time where SSW inlet temperature measurements exceed 75°F is expected to be short.

2. Primary Containment

a) Local Pool Temperature Limits

The containment design limits are based on the ability of the suppression pool to absorb the energy released from the reactor vessel including its inventory with the reactor operating at full power. Technical specification limits on suppression pool temperature ensure a subcooled suppression pool during a LOCA or SRV discharge so that all steam released from the reactor vessel is condensed. The water temperature in the vicinity of the SRV discharge to the suppression pool is referred to in design and licensing documents as the local pool temperature.

Local pool temperature analysis documented in reference 14 was not repeated for a SSW injection temperature of 75°F because recent developments on this aspect of containment design eliminated the need. The following discussion briefly outlines the origin of local pool temperature limits and the current status. FSAR changes are proposed by this safety evaluation to revise sections containing information on local pool temperature limits and analysis.

During plant transients (SORV, LOOP, ATWS, SBO) and small break accidents (PBIC's) requiring SRV actuation, the suppression pool temperature is elevated by steam discharged from the primary system to the suppression pool through the relief valve discharge line. Past experimental data and plant experience indicated the potential for unstable steam condensation during safety-relief valve (SRV) operation at elevated pool temperatures in plants without quenchers. The condensation process in a heated pool causes cyclical pressure loads on the pool boundary and structures inside the wetwell. Suppression pool temperature limits were added to the Technical Specifications defining a safe operating envelope for SRV actuations. These limits in conjunction with plant specific analysis provide assurance that the local pool temperatures during SRV discharge remain within the bounds of experimental data which defined the pressure loads associated with the discharge, and which were used for the design of the containment shell and internal structures.

The local pool temperature limits for SRV discharge are specified in NUREG-0783 [Ref. 15]. FSAR Section 5.2.4.10 "Primary Containment Steam Quenching" includes a summary of the results of analysis performed to demonstrate plant capability to maintain local pool temperatures within the limits contained in NUREG-0783. The highest local pool temperature calculated was 199°F corresponding to a 160°F bulk pool temperature.

Subsequent to NUREG-0783 further experimental data demonstrated that quenchers on the SRV discharge effectively eliminated the previously observed condensation oscillation at higher pool temperature. The NRC issued an SER in August 1994 accepting the conclusions contained in NEDO-30832-A [Ref. 18] regarding "T" quencher performance to mitigate steam condensation loads. On the basis of the experimental evidence provided to the NRC by the BWROG, the local pool temperature limit in NUREG-0783 has been eliminated provided the ECCS pump suction(s) are located below the elevation of the quencher device.

Current Technical Specification limits protect the steam quenching capability of the suppression pool which has an analytical limit for the design basis LOCA of 170°F based on the experimental data from Humboldt and Bodega Bay containment testing. The TS limits take a staggered approach that includes:

- 24 hour limit for bulk temperatures > 80°F ,
- 90°F limit for HPCI, RCIC, and ADS testing,
- 110°F limit before scramming the reactor,
- 120°F limit for reactor vessel depressurization.

The bulk suppression pool temperature limit of 170°F ensures steam quenching capability through the end of vessel blowdown after a LOCA.

The current Technical Specifications require vessel depressurization to below 200 psig at normal cooldown rates when the bulk suppression pool temperature cannot be maintained below 120°F . The calculated change in suppression pool temperature caused by vessel blowdown following a design basis LOCA is less than 50°F . Therefore, by initiating depressurization at 120°F or if the design basis LOCA occurred with the suppression pool at 120°F , the LOCA condensation limit of 170°F will not be exceeded.

The 120°F vessel depressurization requirement (not containment cooling) ensures the 170°F suppression pool temperature steam quenching limit is not exceeded. However, containment cooling is designed to maintain the suppression pool within technical specification limits and return the

suppression pool temperature to < 80°F after testing and within LCO time limits thereby avoiding reactor shutdown and depressurization.

Current Technical Specification Bases 3/4.7.A states:

> "Experimental data indicates that excessive steam condensing loads can be avoided if the peak local temperature of the pressure suppression pool is maintained below 200°F during any period of relief-valve operation with sonic conditions at the discharge exit. Analysis has been performed to verify that the local pool temperature will stay below 200°F and the bulk temperature will stay below 160°F for all SRV transients."

This safety evaluation proposes to replace the above bases to state the following, consistent with the above discussion:

> "The maximum permissible bulk suppression pool temperature of 120°F is permissible since a complete accident blowdown can be accommodated without exceeding the bulk suppression pool temperature limit of 170°F immediately after blowdown."

The proposed bases change removes reference to local pool temperature limits and removes reference to the associated 160°F bulk suppression pool temperature limit.

b) Containment Temperature and Pressure

Analysis of the containment response was performed by General Electric using their proprietary SHEX computer code. This code has been accepted by NRC for calculating the accident and transient containment response.

The model used to calculate the containment response is a coupled reactor and containment model including the systems located outside primary containment that influence the containment response (Feedwater, Core Spray, the RHR system in its various modes, HPCI, etc.). The code performs a fluid mass and energy balance in the reactor primary system, drywell, suppression pool, and wetwell airspace. The code calculations include the time dependent suppression pool bulk temperature, and the pressure and temperature in the drywell and wetwell airspace [Ref. 22].

(1) FSAR Design Basis LOCA Analysis

FSAR Section 14.5.3 contains the accident analysis for the double ended recirculation line break which is the limiting pipe break inside primary containment from a radiological consequence perspective. Of all the postulated transient and accident conditions, the instantaneous double ended circumferential rupture of the recirculation loop suction line represents the most rapid energy transfer from the primary system to the containment and suppression pool. The containment free volume, drywell

vent system capacity, and suppression pool water storage requirements are sized to absorb and suppress this energy release without exceeding the containment design pressure of 56 psig.

The short term response of the containment for a DBA-LOCA is unaffected by the SSW injection temperature since containment cooling is not initiated prior to 600 seconds. For the purposes of this discussion, short term refers to that period of time before operator actions are assumed to occur in the analysis (i.e., t < 600 seconds).

The peak drywell pressure and temperature during a DBA LOCA occurs in the first few seconds of the accident as a result of the reactor blowdown into the containment and is determined by the initial conditions in the reactor vessel, drywell, wetwell, and suppression pool. The short term response of the containment documented in the FSAR remains applicable and was not recalculated for a 75°F SSW injection temperature.

After the initial blowdown and recovery of reactor water level the low pressure ECCS pumps continue to circulate water drawn from the suppression pool through the reactor vessel and core transferring decay and sensible heat back to the pool via the break flow. This phase of recovery from the accident is referred to as the long-term recirculation phase.

Chapter 14 design basis LOCA analysis based on a 65°F SSW inlet temperature assumed that at 600 seconds, one RHR pump is shutoff and the RHR loop (with one pump operating) is placed in suppression pool cooling or suppression pool cooling with spray. The design basis LOCA analysis performed using a 75°F SSW inlet temperature [Ref. 22] assumes a slightly different lineup is used that effectively combines long-term core cooling and containment cooling.

Long-term core and suppression pool cooling can be performed by use of the LPCI with Heat Rejection mode for liquid breaks inside primary containment of sufficient size to support continuous recirculation as described below including the design basis LOCA. In the LPCI with Heat Rejection Mode, the RHRS pumps take suction from the suppression pool, and pump the water through the RHR heat exchanger where cooling takes place by transferring heat to the station cooling water systems. The fluid is then discharged back to the reactor vessel where sensible and decay heat is absorbed. The fluid returns to the suppression pool by flowing out the pipe break, into the drywell, and back to the suppression pool through the drywell to wetwell vent system. This method of cooling sets up a recirculation loop including the suppression pool, RHR heat exchanger, and reactor vessel.

In LPCI with Heat Rejection mode, with two RHR pumps in operation, the heat exchanger bypass valve remains in its full open normal position.

No disruption of LPCI flow is required to enter this mode of suppression pool cooling. This configuration will provide maximum core cooling but does not provide rated heat removal because more than half of the two pump LPCI flow rate goes through the heat exchanger bypass line and not the heat exchanger.

Rated heat removal from the containment is obtained using LPCI with Heat Rejection mode by removal of one RHR pump from LPCI service, closure of the RHR heat exchanger bypass valve, and valve adjustments as necessary to establish 5100 gpm through the RHR heat exchanger."

The design basis LOCA analysis performed using a 75°F SSW inlet temperature [Ref. 22] assumes that at 600 seconds, LPCI with Heat Rejection mode, with two RHR pumps in operation is placed in-service as described above. Two hours after the accident, the accident analysis assumes a transition to LPCI with Heat Rejection using a single RHR pump so that rated heat removal is performed. The single pump configuration of LPCI with Heat Removal is assumed to continue for the entire 30 day duration of the analysis.

The suppression pool will continue to heatup during long-term recirculation and reach its peak temperature approximately 5 to 6 hours after the accident occurred. When sufficient differential temperature is built across the RHR heat exchanger that the heat removal rate exceeds the decay and sensible heat transfer rate into the suppression pool, the suppression pool temperature will start to decrease. The time relative to the beginning of the accident at which the suppression pool peaks and begins declining primarily depends on the time at which containment cooling is initiated and the pump and heat exchanger performance throughout the containment cooling train (i.e., RHR, RBCCW, SSW).

The peak suppression pool temperature obtained for the design basis LOCA is 178°F which is well below the primary containment design temperature limit of 281°F. As discussed above, the peak containment pressure occurs during the initial blowdown before the initiation of containment cooling, and is unchanged from the FSAR value of 45 psig. The effects of the higher suppression pool temperature on the available pump net positive suction head is discussed in a separate section of this evaluation.

(2) FSAR Steam Line Break Analysis

The FSAR steam line break analysis is documented in FSAR Section 5.2.3.2 and based on GE analysis reported in references 52 and 53 which used a SSW injection temperature of 65°F.

The purpose of FSAR steam line break analysis is to evaluate the effect of steam line breaks on the drywell liner which has a design limit of 281°F. A spectrum of small steam line breaks (0.02 to 0.50 ft^2) were evaluated

for effect on drywell liner temperature. Small steam line breaks have the potential to super heat the containment atmosphere up to 330°F and cause the drywell liner to exceed its design temperature. Breaks larger than 0.5 ft^2 involve enough level swell from the reactor vessel that the break flow is a mixture of steam and liquid. The lower energy content of the steam/liquid mixture results in a less severe drywell temperature than for the small break sizes.

Because steam line breaks in this range do not significantly heat the suppression pool prior to the initiation of containment spray, the temperature of the water exiting the RHR heat exchanger is only slightly lower with a 65°F SSW inlet temperature than with a 75°F SSW inlet temperature. In both cases, containment spray effectively reduces the airspace temperature before the containment liner reaches the design temperature limit of 281°F. General Electric has evaluated the FSAR analysis for small steam line breaks considering a SSW injection temperature of 75°F and concluded that the design temperature limit for the drywell liner will not be exceeded [Ref. 20].

(3) Appendix R Fire Events

Appendix R to the Code of Federal Regulations requires that one train of systems necessary to achieve and maintain hot shutdown remain free from fire damage by a single fire. This requirement is necessary to assure the ability to achieve and maintain safe shutdown of the reactor.

The limiting fire event for PNPS is a fire in either Fire Area 1.9 (East Side of Reactor Building 23' Elevation) or 1.10 (West Side of Reactor Building 23' Elevation). These areas are limiting because of the number of plant systems that could be damaged including HPCI, RCIC, a diesel generator, a containment cooling loop, a Core Spray train, a RBCCW loop, a SSW loop, and two SRVs. The key systems that remain available are:

- One Core Spray train
- One RBCCW loop
- One SSW loop
- One RHR Containment Cooling loop
- Two SRVs for manual depressurization
- One diesel generator

Analysis was performed by General Electric [Ref. 24] during implementation of the Appendix R project. The purpose of the analysis was to demonstrate that the required safety functions could be performed with the set of available equipment and to establish minimum operator action times for core cooling and containment cooling [Ref. 26, 27, 28]. This analysis assumed a SSW injection temperature of 65°F and

availability of the systems listed above. This analysis constitutes the analytical basis for fire response procedures 2.4.143.1 and 2.4.143.2.

The above referenced GE analysis predicted that depressurizaton to enable injection with the low pressure core spray system would be required approximately twenty-four minutes after the event initiation. Also, establishment of containment cooling at or before two hours provides an acceptable peak suppression pool temperature that is well below the containment EOP heat capacity temperature limit (HCTL) and the containment design temperature. The availability of sufficient NPSH was also confirmed for the scenario.

General Electric repeated the analysis for the above scenario with the same set of available equipment using a SSW injection temperature of 75°F [Ref. 22]. The time requirement for reactor depressurization remains the same and the results indicate a peak suppression pool temperature of approximately 175°F if suppression pool cooling is initiated two hours after the event. This peak suppression pool temperature is well below the EOP HCTL and well below the containment temperature limit. The availability of sufficient NPSH was also confirmed for the scenario. Based on this new analysis [Ref. 22], the analytical basis previously established for fire response procedures remains valid with a SSW injection temperature of 75°F.

(4) Anticipated Transients Without Scram

In response to the ATWS Rule (10CFR50.62), BECo endorsed GE Licensing Topical Report NEDE-31096-P-A "Anticipated Transients Without Scram" [Ref. 30]. This report specifies design requirements for the Standby Liquid Control (SLC) system, Alternate Rod Insertion (ARI) control logic , and Recirculation Pump Trip (RPT) control logic. These design requirements were evaluated extensively for adequacy in reference 31 using generic plant initial conditions and operating parameters. This analysis demonstrated that critical safety functions for protecting the core, reactor coolant pressure boundary, and containment are met. The design features of PNPS are encompassed by the analysis contained in reference 31. No additional analysis is required to maintain compliance with these generic documents.

Additional plant specific sensitivity analysis for ATWS scenarios is contained in reference 32. This analysis was performed in support of the feedwater pump runback/trip and recirculation pump drive motor trip modifications. The SSW injection temperature used in the analysis is 65°F.

The design verification effort for SSW injection temperature of 75°F included an analysis for the bounding ATWS event evaluated in reference 32. The analysis shows a peak suppression pool temperature of 178°F assuming two containment cooling loops are available [Ref. 22] which is

acceptable under ATWS rules. The results of this analysis demonstrate that containment cooling is adequate for ATWS scenarios with an SSW injection temperature 75°F.

(5) Station Blackout

The station blackout (SBO) rule requires that PNPS be able to withstand and recover from an SBO of 8 hours. Two possible response formats were available for use by licensees in responding to the rule: (1) use of an Alternate AC (AAC) power source, or (2) a response independent of AC power. BECo chose to use the first response alternative because of the availability of the SBO diesel. Option 2 would have required a detailed analysis of plant response including containment temperatures, and building temperatures.

The non-Class 1E SBO diesel is relied on as a source of power following a SBO. This diesel can be made available within 10 minutes of the event and is capable of providing power for all equipment necessary to perform containment cooling, component cooling, and building compartment cooling [Ref. 8].

Because of the availability of the SBO diesel generator, the plant response to a SBO will be similar to an isolation event (e.g. Loss-of-Offsite-Power). Since, the core cooling, containment cooling, component cooling, and building compartment cooling requirements for a LOOP are much less severe than those for a design basis LOCA, the above results for a design basis LOCA bound the results from an SBO analysis. Based on this information cooling related safety functions during a SBO can be accomplished at a SSW injection temperature 75°F.

(6) Containment Metal-Water Reaction Capability

Metal-water reaction can occur when zircaloy is heated above a temperature of 2000°F in the presence of steam. The products of this type of exothermic reaction are zirconium oxide, hydrogen, and energy release of about 2800 BTU/lb of reacted zirconium.

The hydrogen produced will add to the mass of noncondensible gas already in the containment resulting in an increased containment pressure. The energy released from the reaction will be absorbed in the suppression pool raising its temperature. The containment capability to accommodate metal-water reaction is partially dependent on containment cooling and that capability far exceeds the metal-water reaction which is expected to occur.

The containment capability with respect to metal-water reaction is defined as the maximum percentage of fuel channels and cladding material that can enter a metal-water reaction without exceeding the maximum allowable containment pressure. FSAR Section 14.5.3.1.4 establishes the

metal-water reaction capability of the containment at 20% using a SSW injection temperature of 65°F [Ref. 13a].

General Electric approximates the reduction in containment metal-water reaction capability due to a 10°F increase in SSW injection temperature at less than 1%. Therefore, the containment metal water reaction capability at a SSW injection temperature of 75°F is approximately 19% which is well above the calculated metal water reaction of < 0.1%. [Ref. 22 & 23].

3. ECCS NPSH

Net positive suction head (NPSH) availability exceeding the requirement for each pump is critical to ensure smooth reliable pump operation throughout the accident or transient response. Net positive suction head available (NPSHA) at the pump inlet is equal to the total absolute suction pressure minus the vapor pressure of water at the suppression pool temperature. The NPSH required (NPSHR) at the pump suction is the minimum pressure over and above the vapor pressure that must be present in order to keep the water in the liquid state as it enters the impeller. NPSHR is provided by the pump manufacturer and based on testing of individual pumps.

The design basis LOCA is the bounding accident for NPSH analysis because it results in the highest bulk suppression pool temperature coupled with the lowest containment pressure and, hence, the least NPSH margin (e.g., NPSHA minus NPSHR). The design basis for ECCS pump NPSHA was evaluated for the design basis LOCA using a SSW inlet temperature of 65°F in Safety Evaluation 2971. This evaluation will not repeat the detailed technical, licensing and design basis information to the extent provided in that safety evaluation. Rather, this evaluation will provide the results of the NPSH evaluation that was performed for the design basis LOCA using a SSW injection temperature of 75°F and compare those results against the same acceptance criteria as SE2971.

A 10°F SSW injection temperature increase will result in a higher suppression pool temperature over the course of the event than would otherwise occur at the lower SSW injection temperature. General Electric has evaluated the design basis LOCA including the suppression pool temperature response [Ref. 22] using a 75°F SSW inlet temperature. Using this design basis LOCA suppression pool response, NPSH available was calculated using the same methodology and inputs presented in SE2971 including updated values for pump NPSHR, suction line loss, containment initial conditions, and containment noncondensible gas leakage.

The results of the calculation are included with this safety evaluation as new FSAR Figures 14.5-18, and 19 [Ref. 41]. The new figures also show that containment pressure does not continue to decrease below

atmospheric pressure since this is not possible but was depicted as such on the original FSAR figures. In addition, the design value for containment leakage is now included as 1% per day at a reference pressure of 45 psig to be consistent with Tech Spec 4.7.A.2, and at a degraded leakage rate of 5%.

NPSH margin with leakage effects included, is presented on Figure 14.5-18 and 19. Here, the suppression pool temperature and containment pressure are shown as a function of time. Also shown is the primary containment pressure required to provide the required NPSH to the RHR and Core Spray pumps at their maximum required flow rates. As can be seen, substantial margin exists throughout the duration of the event. Therefore, it can be concluded that adequate NPSH will be available at all times following a design basis LOCA.

The potential effect from insulation debris accumulating on ECCS pump suction strainers has been evaluated [Ref. 41]. The conclusion is that the increase in suction head loss from the postulated debris accumulation is within the margin for NPSH available to the ECCS pumps. Since the NPSH available at the pump suction exceeds the NPSH required, the pump will achieve its rated performance. Therefore, there is no effect on ECCS pump performance and no change in the margin of safety as determined by the accident analyses. Furthermore, the assumptions made in this evaluation are consistent with the bases used for the Technical Specification requirements applicable to the Containment Systems (Section 3.7.A).

The RHR and Core Spray system design analysis shows positive NPSH margin is available following the bounding design basis LOCA without containment positive pressurization beyond the initial pressurization that can be obtained from the nitrogen gas stored in the drywell and wetwell airspace. The design margin for NPSH available is that which exists between the minimum containment pressure that provides the required NPSH and the containment pressure that exists due to equilibrium conditions for the gas/vapor mixture with an accounting for containment initial conditions and leakage.

4. **Higher Suppression Pool Temperature**

Although a 75°F SSW injection temperature results in a higher peak suppression pool temperature than in the case of a 65°F SSW injection temperature, the time of occurrence of the peak pool temperature in both cases roughly coincides (approximately 5 to 6 hours after the start of the accident). By the time suppression pool temperature exceeds those predicted by the 65°F analysis, core decay heat and temperature is well below the critical range (i.e. PCT approaching the limit) and the difference

in pool temperature between the two cases (12°F at the peak) will have a negligible effect on core cooling.

Furthermore, analysis performed by the BWROG to raise the initial temperature of suppression pools evaluated the effect of higher suppression pool temperature on core standby cooling system performance. It was determined based on available analysis that a 50°F increase in the temperature of water injected into the reactor for core cooling would result in approximately a 6°F increase in PCT. This evaluation assumed the higher temperature water was used early in the event when cooling is most critical [Ref. 17].

5. Component Cooling

For ultimate heat sink temperatures up to and including 65°F, heat exchangers and equipment performance criteria is based on a design basis LOCA peak suppression pool temperature of 166°F. For example, fouling factors, and tube plugging limits are based on the 65°F design case. For heat sink temperatures above 65°F, the design basis peak suppression pool temperature increases up to a peak of 178°F with a 75°F SSW inlet temperature.

a) Residual Heat Removal System

(1) RHR Heat Exchanger

The current FSAR analysis for the limiting DBA-LOCA event predicts a peak suppression pool temperature of 166°F using a SSW injection temperature of 65°F. At these conditions, the instantaneous heat transfer rate for the RHR heat exchanger is 64×10^6 BTU/hr. Analysis of the limiting DBA-LOCA event using a SSW injection temperature of 75°F predicts a peak suppression pool temperature of 178°F. At these conditions the instantaneous heat transfer rate for the RHR heat exchanger is 65.1×10^6 BTU/hr. The peak suppression pool temperature of 178°F is far less than the containment design temperature of 281°F. Therefore, the performance of RHR heat exchanger for the limiting SSW injection temperature of 75°F is adequate.

(2) Shutdown Cooling Capability

FSAR Section 4.8.4 states that the SDC subsystem is capable of maintaining reactor water temperatures at 125°F approximately 20 hours after reactor shutdown so that the reactor can be refueled and serviced. This capability was based on a SSW injection temperature of 55°F [Ref. 2a]. Also, the NRC in Regulatory Guide 1.139 recommends that cold shutdown be achieved within 36 hours after reactor shutdown with a single failure [Ref. 19].

An analysis was performed with the RHR system aligned in SDC mode to evaluate decay heat removal capability with an SSW injection temperature of 75°F . The shutdown analysis assumes a maximum cooldown rate of 50°F/hr and the SDC subsystem is initiated with a moderator temperature of 260°F at approximately 6 hours after shutdown. With both RHR heat exchangers in operation along with the associated cooling trains, the reactor water temperature can be reduced to 125°F within 17 hours after reactor shutdown [Ref. 20].

If a single cooling train (one RHR heat exchanger) of SDC is available, the reactor coolant can be cooled to below 212°F within 8 hours after shutdown and to 150°F within 33 hours after shutdown [Ref. 20]. These analysis results demonstrate SDC capability consistent with current FSAR commitments as well as the recommendations in Regulatory Guide 1.139.

b) **RBCCW System**

(1) RBCCW Heat Exchanger

Under emergency conditions, each RBCCW heat exchanger transfers heat from the RHR heat exchanger, Equipment Area Coolers, RHR Pump Seal Coolers, and CS Thrust Bearing Cooler to the ultimate heat sink via the SSW system. The limiting event for heat transfer requirements is a DBA-LOCA with a failure of one division of containment cooling. In this event the containment cooling safety function is performed by one loop of containment cooling. The conditions associated with a DBA-LOCA cause the most rapid transfer of decay and sensible heat from the reactor to the containment and suppression pool. Accordingly, the highest predicted suppression pool temperature out of all accident and transient scenarios results from these event assumptions.

The current FSAR analysis for the limiting DBA-LOCA event predicts a peak suppression pool temperature of 166°F using a SSW injection temperature of 65°F. At these conditions, the instantaneous heat transfer rate for the RBCCW heat exchanger is 65×10^6 BTU/hr. Analysis of the limiting DBA-LOCA event using a SSW injection temperature of 75°F predicts a peak suppression pool temperature of 178°F. At these conditions the instantaneous heat transfer rate for the RBCCW heat exchanger is 66.1×10^6 BTU/hr. The peak suppression pool temperature of 178°F is far less than the containment design temperature of 281°F. Therefore, the performance of RBCCW for the limiting SSW injection temperature of 75°F is adequate.

During a design basis LOCA with a SSW injection temperature of 75°F, a peak cooling water inlet temperature of 98°F to the area and component coolers coincides with the occurrence of the peak

suppression pool temperature of 178°F. This cooling water inlet temperature is evaluated below.

(2) RHR Pump Seal Coolers

Each RHR pump is equipped with a seal cooler which lowers the temperature of the water injected into the seal chamber to flush and cool the rotating seal faces so that there will not be flashing at the seal faces. By maintaining the temperature of the seal chamber below saturation, flashing at the seal face is prevented. The seal cooler is necessary because the RHR pumps are used for the SDC mode of the RHR system where water temperature can exceed 300°F. The seal cooler is not essential for LPCI or containment cooling since the temperature of the fluid handled is well below the saturation temperature. Consistent with this basis the Core Spray pumps have a similar mechanical seal and do not have a seal cooler since the fluid handled by these pumps is either relatively cool water from the CST's or suppression pool water at the same temperature as that handled by the RHR pumps during transient and accident scenarios.

The peak suppression pool temperature calculated for a loss of coolant accident is 178°F using a SSW injection temperature of 75°F, well below the saturation temperature. Although the seal cooler is not considered essential for the accident case, RBCCW coolant passes through the seal cooler and a small amount of additional energy is transferred to the RBCCW loop and removed via the RBCCW heat exchanger. This small contribution to the heat load on the RBCCW heat exchanger is evaluated in cooling loop performance calculations.

(3) Core Spray Pump Motor Cooling

The Core Spray pumps do not include a cooler for its mechanical seal because the temperature of the fluid handled by the pump will not result in flashing at the seal faces. Each Core Spray pump motor has a cooling coil immersed in the oil bath provided to lubricate the motor thrust bearings. The calculated cooling water inlet and outlet temperatures are well within the acceptable range to ensure reliable operation of the Core Spray pump motors with a SSW injection temperature of 75°F [Ref. 43].

RBCCW coolant passes through the thrust bearing(s) cooler and a small amount of additional energy is transferred to the RBCCW loop and removed via the RBCCW heat exchanger. This small contribution to the heat load on the RBCCW heat exchanger is evaluated in cooling loop performance calculations.

6. Building Compartment Cooling

a) Equipment Area Cooling

Safety-related area coolers are provided in the northwest ("B Quad"), southeast ("A Quad"), southwest ("RCIC Quad"), and HPCI pump room. The area coolers are required to assist in maintaining these areas at relatively low temperature to ensure reliable operation of the local equipment.

Area cooler performance is rated with a 85°F inlet cooling water temperature. During a design basis LOCA with a SSW injection temperature of 75°F, a peak cooling water inlet temperature of 98°F to the area coolers coincides with the occurrence of the peak suppression pool temperature of 178°F. Therefore, a performance analysis for the equipment area coolers calculated a "K" factor used to represent air cooler performance at a given flow rate for a range of inlet temperatures from 85°F up to 100°F [Ref. 44]. These "K" factors were inputs to the analysis used to determine the post-accident area temperatures in the secondary containment [Ref. 37]. The temperature results for building compartments served by equipment area cooling as well as those that do not were evaluated against temperature limits for the local equipment and found acceptable to ensure reliable operation.

b) Motor Control Center Ventilation

Assuming a SSW inlet temperature of 65°F, the temperature in both the ventilated and unventilated motor control center (MCC) enclosures had been evaluated previously for the temperature environment on the 23 foot elevation of the Reactor Building following a design basis LOCA and found to be acceptable [Ref. 38]. As part of the design verification effort, these MCC enclosures were evaluated assuming a SSW inlet temperature of 75°F. The temperature in the enclosures will remain sufficiently low to ensure reliable operation of the motor control centers [Ref. 39].

7. Environmental Qualification

Environmental qualification (EQ) is required by the Code of Federal Regulations (10CFR50.49) for equipment necessary to ensure health and safety of the public during postulated accidents. Equipment at PNPS requiring qualification to meet the requirements of 10CFR50.49 are listed in the Environmental Qualification Master List [Ref. 46].

The environmental conditions (e.g., radiation, pressure, temperature, humidity etc.) for which equipment listed on the EQML must be qualified are contained in specification E536 broken down by plant area (e.g. building compartment, drywell, wetwell etc.). Some qualification parameters specified in E536 [Ref. 47] are affected by the increase in SSW injection temperature. The affected parameters were re-evaluated for a 75°F SSW injection temperature [Ref. 48] and are discussed individually below.

a) Drywell Temperature

The worst case event for drywell airspace temperature is a small steam line break of approximately 0.5 ft^2. Breaks of this size and type cause a superheated steam environment in the drywell airspace with airspace temperature exceeding 330°F. The range of small breaks analyzed to establish the worst case drywell airspace temperature profile is 0.01 ft^2 to 1.0 ft^2. Breaks larger than 0.5 ft^2 involve enough level swell from the reactor vessel that the break flow is a mixture of steam and liquid. The lower energy content of the steam/liquid mixture results in a less severe drywell temperature than for the small break sizes.

Figure C.4.1-18 in specification E536 contains a plot of the drywell airspace temperature qualification requirement as a function of time after the accident. This curve envelopes the temperatures calculated for a spectrum of small steam line breaks (0.01 ft^2 to 1.0 ft^2) analyzed for their effect on drywell temperature. Figure C.4.1-18 in reference 47 is based on analysis contained in reference 49. That analysis was based on a SSW injection temperature of 65°F.

Drywell airspace temperature analysis was performed by GE using a SSW inlet temperature of 75°F [Ref. 22]. The FSAR analysis described previously is different in two respects from the steam line break analysis for environmental qualification as tabulated below:

Input Description	EQ Analysis	FSAR Analysis	Justification
Passive heat sinks in the drywell, wetwell, and suppression pool modeled	Yes	No	Realistic heat transfer taken credit for in EQ analysis.
Drywell Spray Flow Rate	720	300	Realistic flow rate for EQ analysis

These differences provide a more realistic estimate of the drywell temperature response than the FSAR analysis case. The passive heat sinks tend to moderate the drywell temperature response by absorbing energy from the break during drywell heatup and releasing energy later during drywell cooldown. The higher spray flow rate used for EQ analysis provides more cooling of the drywell airspace and reduced temperature for equipment located in the drywell.

The resulting qualification envelope developed from the GE analysis is more severe than that provided in 1987 by reference 49 and much more severe than would be expected for a 10°F SSW injection temperature increase. This discrepancy was documented in PR96.9028 [Ref. 50].

A review and update of all environmental qualification data files was performed to verify environmental qualification at the new 75°F qualification envelope for drywell temperature. Existing qualification test documentation

is sufficient to demonstrate qualification of most affected components [Ref. 54 & 55]. However, additional information was required to complete qualification document files for five motor operated valves located inside the drywell and the containment electrical penetration assemblies.

Additional qualification testing for subcomponents inside the five MOVs has been completed at Wyle Laboratories. Results of the testing [Ref. 51] conclude that the MOV subcomponents are qualified to the drywell temperature profile based on a 75°F SSW inlet temperature.

b) Containment Penetrations

Safety related equipment(s) located inside primary containment that must operate post accident depends upon power, signal, or control cabling routed through containment electrical penetrations. FSAR Sections 5.2.3.4.3 and 7.1.7 summarize the physical characteristics and capabilities of the various penetration designs under post accident conditions. Low voltage power and control penetration designs have an inner and outer seal made of epoxy resin potting compound surrounding each individual conductor. This resin is relied on to maintain the penetration leaktight under worst case LOCA conditions to ensure containment integrity. Furthermore, when essential equipment is powered by cabling routed through a penetration, the containment penetration and the associated cabling must support the operation of the associated electrical equipment for its entire post accident mission time.

The containment electrical penetrations that support the operation of active equipment are listed on the EQ Master List [Ref. 46]. Containment electrical penetrations not listed on the EQ Master List contain cabling that is not required to function electrically in a post accident environment but may continue to operate. Although functionally passive, penetrations not listed on the EQ Master List must remain leaktight to ensure containment integrity.

A review and update of all environmental qualification data files for containment penetrations listed on the EQ Master List was performed and all penetrations listed on the EQ Master List are qualified at the new 75°F qualification envelope for drywell temperature.

The electrical penetrations not listed on the EQ Master List were tested during initial qualification testing to accident conditions that exceed the predicted accident conditions for both steam and liquid line breaks.

c) Protection of Electrical Penetrations

The possibility of an electrical fault (i.e., short circuit) or overload in electrical circuits routed in the drywell is credible during a LOCA because of the harsh environment caused by high energy steam and water discharge from the reactor vessel. Electrical faults and overloads will cause higher than normal current flow in the conductor with an accompanying temperature rise.

The resulting temperature of cables routed through containment penetrations that experience overload or fault currents must be protected by devices such that containment integrity is maintained. FSAR Section 8.9.5 "Cable Protection and Process Instrumentation Location Criteria" states the power cables are derated according to Insulated Power Cable Engineers Association (IPCEA) procedures depending on factors that include ambient temperature, fault current, and spacing (i.e., air gap between adjacent cables). Also, the FSAR states that overload protection is provided by the proper selection and setting of relays, circuit breakers, heaters, and fuses. Proper implementation of these design requirements for cables routed through containment penetrations is particularly critical because if a overload or fault occurs, containment integrity is partially dependent on this protection.

Problem Report 96.9092 [Ref. 59] identifies deficiencies found in the fault and overload protection arrangements for particular cables routed through containment electrical penetrations. These problems are a result of existing cable size selections for particular loads, breaker settings, and thermal overload relay trip settings. An operability evaluation using NEDWI 395 guidance has been prepared for each of the problem circuits.

This particular problem is discussed in this evaluation in order to document that the particular cable overheating problems identified in PR96.9092 are in part caused by the temperatures from the initial reactor blowdown into the primary containment. Since the containment airspace temperature associated with the initial blowdown is not influenced by the SSW inlet temperature, the identified cable overheating problems are independent of the ultimate heat sink temperature.

d) Suppression Pool Temperature

The DBA-LOCA suppression pool profile includes the highest peak temperature, most rapid pool heatup, and highest long-term temperature when compared against all other design basis events. However, no equipment listed on the EQML is submerged in the suppression pool or located inside the containment wetwell. Therefore, the increased suppression pool temperature does not directly affect qualification requirements for any equipment listed on the EQML.

e) Building Compartment Temperatures

The ambient temperature in building compartments throughout the secondary containment following a DBA-LOCA are primarily a function of the initial temperature, heat gain due to process heat loads and electrical heat loads, active cooling systems (e.g. area cooling, and ventilation), and heat loss to the environs by heat transfer through building walls, floors, and ceilings. Building compartment temperature profiles for important areas of the plant [Ref. 36] were based on containment heat loads (i.e., drywell temperature profile, suppression pool temperature profile) and piping process loads based

on a SSW injection temperature of 65°F. These building compartment temperature profiles represent the post LOCA ambient temperature requirement for equipment located in the secondary containment.

The design verification effort included development of building temperature profiles based on a SSW injection temperature of 75°F. Appropriate containment heat loads and process heat loads were utilized in the analysis. Equipment qualification was verified for resulting post-LOCA building ambient temperature profiles [Ref. 58].

8. **Standby Gas Treatment**

The Standby Gas Treatment System (SBGTS) is part of the secondary containment system and in conjunction with the primary containment and other emergency safeguards acts to minimize the release of radioactive material from an accident so that the offsite dose will be below the guideline values stated in 10CFR100. The SBGTS maintains the secondary containment at a negative pressure relative to the environs to prevent a ground level release of radioactive material. The SBGTS takes a suction from the reactor building, drawing the effluent through a high efficiency particulate absorber (HEPA), two charcoal beds and a final HEPA filter and discharges the effluent to the environs via an elevated release point.

The maximum allowable temperature of the building effluent entering the SBGTS is 200°F [Ref. 56 & 57] based on the established set point for temperature switches in the heater control circuit. The predicted maximum effluent temperature entering the SBGTS during the design basis LOCA is below 120°F based on secondary containment heatup analysis using a 65°F SSW inlet temperature which is well below the limit. A ten degree rise in the SSW inlet temperature will not cause the maximum allowable effluent temperature to be exceeded. Therefore, continuous operation of the SBGTS is ensured following a design basis LOCA, since air temperature in building compartments from which the SBGTS takes suction is below the temperature limit that causes the SBGTS train(s) to automatically shut down.

9. **Standby AC Power Source**

In the event of a single failure that causes a loss of one diesel generator set, the remaining diesel generator is required to provided sufficient power to its associated emergency bus to ensure the performance of core, containment, component, and building compartment cooling as required in the accident analysis.

Assumptions from the Chapter 14 design basis LOCA analysis for PNPS at 65°F SSW inlet temperature are reflected in the diesel loading (FSAR Table 8.5-1) as follows:

- 10 minutes after the accident one RHR pump is shutoff and one additional SSW and RBCCW pump are started to support containment cooling initiation.

The analysis performed for the design basis LOCA using a 75°F SSW injection temperature assumed a different containment cooling method than was assumed in the above analysis. This method combines core and containment cooling in a mode designated LPCI with Heat Rejection and causes an increased diesel generator load for a period of time during the accident response.

Design Basis LOCA 65°F SSW Injection Temperature	
0 < t < 10 minutes [KW]	t > 10 minutes [KW]
1 CS Pump [639] 2 LPCI Pumps [1278] 1 SSW Pump [83] 1 RBCCW Pump [54]	1 CS Pump [639] 1 RHR - SPC Pump [639] 2 SSW Pumps [166] 2 RBCCW Pump [108]
2,054 KW	1,552 KW

Design Basis LOCA 75°F SSW Injection Temperature		
0 < t < 10 minutes [KW]	10 min < t <2 hours [KW]	t > 2 hours [KW]
1 CS Pump [639] 2 LPCI Pumps [1,278] 1 SSW Pump [83] 1 RBCCW Pump [54]	1 CS Pump [639] 2 LPCI Pumps [1,278] 2 SSW Pumps [166] 2 RBCCW Pumps [108]	1 CS Pump [639] 1 LPCI Pumps [639] 2 SSW Pumps [166] 2 RBCCW Pump [108]
2,054 KW	2,191 KW	1,552 KW

The bottom row of each of the above tables contains the total kilowatt hour load associated with the listed pumps. The total load is the same for the first 10 minutes in both analysis. However, the 75°F case has a higher load than the 65°F case in the period after the accident between 10 minutes and 2 hours. For that one hour and fifty minute period of time the diesel load is greater than previously analyzed by 639 KW because an additional RHR pump is assumed to operate.

A review of the available margin for additional diesel load as tabulated in calculation PS79 [Ref. 60, 61], indicates that the total diesel load will remain within the 2000 hour capacity of 2,750 KW per Table 8.5-3 assuming no failures. According to the calculation, the most heavily loaded diesel is "B" with a steady-state load of 2,563 KW in the period of time between accident initiation and 10 minutes. Assuming the additional SSW and RBCCW pumps are started at ten minutes after essential valves have cycled, the diesel load is 2,700 KW which is below the 2,000 hour limit of 2,750 KW.

Based on the above the most heavily loaded diesel generator will not be overloaded while performing core and containment cooling as assumed in the design basis LOCA analysis using a SSW inlet temperature of 75°F.

Reference 60 considers individual single failures (e.g., load shed logic failure, or salt service water pressure switch failure) that can cause an overload condition on one diesel generator. However, the remaining diesel generator is capable of performing the required safety functions of the standby AC power system. The possibility of a failure of one diesel generator is assumed when evaluating the integrated core and containment cooling system(s) response to design basis events and this failure will not cause any design limit associated with the radioactive material barriers to be exceeded.

10. Piping Design

If an accident or transient were to occur with the SSW injection temperature above 65°F, the temperature of fluid circulated by RHR, Core Spray, RBCCW, SSW systems would be higher for a period of time during the accident response than would be expected for a 65°F SSW inlet temperature. The DBA-LOCA causes the highest predicted temperature for the fluids circulated in these piping systems [Ref. 40]. The piping analyses for these systems is based on a design temperature that could be exceeded for a period of time during the event response. Analysis of the effect of the increased piping temperature caused by higher fluid temperature was performed to ensure that changes to piping or support loads and stresses were identified and analyzed.

Based on evaluations documented in reference 45 the current configuration of piping and supports in the above systems is acceptable for a 75°F SSW injection temperature. This conclusion is based on the following:

- The existing piping analyses is based on a design temperature that exceeds the expected temperature during the design basis event response (e.g., parts of the RHR system that are used for SDC mode, and parts of the RHR and CS systems between the first and second isolation valves at the RCPB).
- The piping temperature is so low (e.g., near ambient temperature for SSW inlet piping) that stresses related to thermal conditions are minimal.
- Moderate temperature increases are predicted for the SSW discharge piping, Core Spray piping, and RBCCW piping. The piping design in these systems is dominated by weight and seismic loads and the additional thermal effects anticipated are easily tolerated with no significant overall effect on the piping, supports, or components.

H. Summary

1. Q: May the proposed activity increase the probability of occurrence of an accident previously evaluated in the Final Safety Analysis Report?

   A: No, the site maximum SSW inlet temperature of 75°F does not change or influence factors that might cause an accident evaluated in the FSAR to occur.

   Furthermore, the plant was designed to operate normally with a seawater temperature of 75°F. Therefore, operation at 75°F seawater temperature will not increase the probability of occurrence of transients (i.e., loss of condenser vacuum) that might challenge safety related systems.

2. Q: May the proposed activity increase the consequences of an accident previously evaluated in the Final Safety Analysis Report?

   A: No, at the site maximum SSW inlet temperature the core, containment, component, and building compartment cooling systems are capable of meeting all performance requirements necessary to ensure physical barriers against the release of radioactive materials perform as intended to limit offsite dose consequences to those previously evaluated in the FSAR.

3. Q; May the proposed activity increase the probability of occurrence of a malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report?

   A: No, although the site maximum SSW inlet temperature of 75°F results in higher temperatures in the plant than would be seen at the design value of 65°F, systems, structures and components important to safety can reliably perform as intended during each of the design basis events.

4. Q: May the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the Final Safety Analysis Report?

   A: No, at the site maximum SSW inlet temperature the core, containment, component, and building compartment cooling systems are capable of meeting all performance requirements necessary to ensure physical barriers against the release of radioactive materials perform as intended to limit offsite dose consequences to those previously evaluated in the FSAR.

5. Q: May the proposed activity create the possibility of an accident of a different type than any previously evaluated in the Final Safety Analysis Report?

   A: No, there are no new or accident initiators or failures resulting from a SSW inlet temperature of 75°F versus 65°F. The types and basic nature of accidents associated with the station design is unchanged.

6. Q: May the proposed activity create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated in the Final Safety Analysis Report?

   A: No, although the site maximum SSW inlet temperature of 75°F results in higher temperatures and pressures in the plant than would be seen at the design value of 65°F, there are no new or different types of equipment malfunction caused by these changes.

7. Q: Does the proposed activity reduce the margin of safety as defined in the basis for any Technical Specification?

   A: No, Technical Specification Bases 3.5.B "Containment Cooling" states that "Each system has the capability to perform its function; (i.e., removing 64 x 10^6 Btu/hr (Ref. Amendment 18), even with some system degradation." This heat removal capability refers to the containment cooling heat load from the RHR heat exchanger and is based on the SSW inlet temperature design value of 65°F. In addition to the containment heat load, the closed cooling water loop also removes approximately 1 x 10^6 Btu/hr from other essential equipment during a design basis LOCA.

   Using FSAR site maximum ultimate heat sink temperature of 75°F for the SSW inlet temperature, each system has the capability to remove 65.1 x 10^6 Btu/hr with some degradation. In addition to the containment heat load, the closed cooling water loop also removes approximately 1 x 10^6 Btu/hr from other essential equipment during a design basis LOCA.

   Therefore, this change does not reduce the margin of safety in the basis for any technical specification.

I. References

(1) FSAR Section 2.4 "Hydrology

(2) FSAR Section 4.8 "Residual Heat Removal System"

(a) FSAR Table 4.8-1 "Residual Heat Removal System Equipment Design Data"

(3) FSAR Section 5.2 "Primary Containment System"

(4) FSAR Section 5.3 "Secondary Containment System"

(5) FSAR Section 6.4 "Core Standby Cooling System"

(6) FSAR Section 7

(7) FSAR Section 7.1.7 "Safety Related Components Inside Containment: Environmental Qualification"

(8) FSAR Section 8

(9) FSAR Section 10.5 "Reactor Building Closed Cooling Water System"

(10) FSAR Section 10.7 "Salt Service Water System"

(11) FSAR Section 10.18 "Equipment Area Cooling System"

(12) FSAR Section 14.5 "Transient and Accident Analysis"

(13) FSAR Section 14.5.3 "Loss of Coolant Accident"

(a) FSAR Section 14.5.3.1.4 "Metal Water Reaction Effects on Primary Containment"

(14) General Electric Company, "PNPS Unit 1 Suppression Pool Temperature Response, " NEDC-22089P, March 1982.

(15) Suppression Pool Temperature Limits for BWR Containment's, NUREG-0783, November 1981.

(16) "Transmittal of the GE Topical Reports on Suppression Pool Temperature Limit, NEDO-30932 and NEDO-31695", BWROG-95067, Letter from BWROG Chairman R.A. Pinelli to BWROG Primary Representatives of Participating Utilities, August 24, 1995.

(17) General Electric Company, "BWR Suppression Pool Temperature Technical Specification Limits," NEDO-31695-A, May 1995.

(18) General Electric Company, "Elimination of Limit on BWR Suppression Pool Temperature for SRV Discharge with Quenchers," NEDO-30832-A, May 1995.

(19) USNRC Regulatory Guide 1.139, Guidance for Residual Heat Removal, May 1978.

(20) GE-NE-523-A044-0595, "Pilgrim Nuclear Power Station Decay Heat Removal Capability", May 1995, (SUDDS/RF95-127, Rev. 1)

(21) Not Used

(22) GE-NE-T23-00732-01, "Pilgrim Nuclear Power Station Containment Heat Removal Analysis," March 1996, (SUDDS/RF96-05, Rev. 0)

(23) NEDC-31852P, "Pilgrim Nuclear Power Station SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," April 1992.

(24) EAS82-0787, "Safe Shutdown Appendix R Analyses for the Pilgrim Nuclear Power Plant", (SUDDS/RF87-889).

(25) MDE59-0486, "Safe Shutdown Appendix R Analyses for Fire Event at One Shutdown Panel for Pilgrim Nuclear Power Station," (SUDDS/RF86-42).

(26) SE2229, "Add Fuses to RBCCW/SSW Alternate Shutdown Panels and Reroute Control Cables for MO1400-25B", dated 10/14/87.

(27) SE2270, "Procedure 2.4.143.1 Shutdown with a Fire in Reactor Building East (Fire Area 1.9) Rev. 0", dated 4/22/88.

(28) SE2271, "Procedure 2.4.143.2 Shutdown with a Fire in Reactor Building West (Fire Area 1.10) Rev. 0", dated 4/22/88.

(29) SE2971, "Replace all piping thermal insulation in the drywell with Owens-Corning NUKON fiberglass blanket insulation".

(30) NEDE-31096-P-A, "Anticipated Transients Without Scram, Response to NRC ATWS Rule 10CFR50.62," February 1987.

(31) NEDE-24222, "Assessment of BWR Mitigation of ATWS, Volume II (NUREG 0460 Alternate No. 3)," December 1979.

(32) NEDC-31425, "Evaluation of ATWS Performance at Pilgrim Nuclear Power Station," September 1989.

(33) BECo Calculation S&SA90, "Ultimate Heat Sink Temperature Analysis", Rev. 0, 3/7/96.

(34) BECo Calculation S&SA91, "Containment and Decay Heat Removal Analysis Inputs", Rev E1, 3/21/96

(35) BECo Calculation S&SA94, "Develop Drywell Composite Temperature Curve from Steam Line Break Analysis Results," Rev. E0, 3/25/96.

(36) BECo Calculation N127, "Secondary Containment Heatup after LOCA without LOOP" Rev. E0, 12/2/93

(37) BECo Calculation N127, "Secondary Containment Heatup after LOCA without LOOP" Rev. E1, 3/25/96

(38) BECo Calculation N142, "MCC Enclosure Temperatures" Rev. E0, 12/15/93

(39) BECo Calculation N142, "MCC Enclosure Temperatures" Rev. 1, 3/25/96.

(40) BECo Calculation M641, "RBCCW Heat Exchanger Performance", Rev E0, 3/25/96.

(41) BECo Calculation M662, "RHR and Core Spray Pump NPSH and Suction Pressure Drop," Rev. E1, 3/25/96.

(42) BECo Calculation M 663, "RHR Heat Exchanger Performance", Rev. E0, 3/25/96.

(43) BECo Calculation M 664, "Containment Heat Removal", Rev. E0, 3/25/96.

(44) BECo Calculation M 665, "Equipment Area Cooler Performance", Rev. E0, 3/25/96.

(45) BECo Calculation M 673, "Evaluation/Reconciliation of Increased Piping Temperatures", Rev. 0, 3/17/96.

(46) Boston Edison Co., "Environmental Qualification Master List", Rev. E44, 11/15/95.

(47) BECo Specification E536, "Environmental Parameters for Use in the Equipment Qualification of Electrical Equipment (per 10CFR50.49)," Rev 4.

(48) BECo Specification E536, "Environmental Parameters for Use in the Equipment Qualification of Electrical Equipment (per 10CFR50.49)," Rev 5.

(49) EAS98-0887, "Drywell Temperature Analysis," General Electric Company, August 1987, (SUDDS/RF87-917)

(50) PR96.9028, "Questions Raised with GE Preliminary Data Tables for 75°F SSW Injection Temperature", 1/29/96.

(51) EQDF 425, Wyle Test Report.

(52) EAS52-0587, "Impact on Containment Pressure Temperature Response of Proposed Capping of Certain Drywell Spray Sparger Nozzles," Rev.1, General Electric Company, May 1987, (SUDDS/RF87-825).

(53) EAS56-0888, "Information on the Effect of Reduced Drywell Spray Flow Rates," General Electric Company, September 1988, (SUDDS/RF88-187).

(54) ICED96-012, "LOCA without LOOP Closure of All Documentation Reviews," dated 3/18/96.

(55) EED96-~~024~~, "Qualification of EDD Equipment at 75°F SSW Injection Temperature".

(56) Calculation I-N1-197, "Standby Gas Treatment System Heater Control Set Point Calculation, TS81000A & B", Rev. 0.

(57) Functional Description SM437 Sheet 4, "Standby Gas Treatment System", Rev. E5.

(58) Equipment Qualification Data Files 420 and 420A, "LOCA without LOOP Temperature Profile Evaluation.

(59) Problem Report 96.9092, Problems Identified in NEDWI 395 Operability Evaluation Dated 1/26/95 for the Primary Containment Electrical Penentrations.

(60) BECo Calculation PS79, "Diesel Generator Loading", Rev. 4.

FSAR Changes

FSAR changes are identified below and on the attached FSAR page markups.

1. **Replace existing title "Section 1.6.2.12 Residual Heat Removal System (Containment Spray and Suppression Pool Cooling)" with the following:**

   Section 1.6.2.12 Residual Heat Removal System (Containment Spray, Suppression Pool Cooling, LPCI with Heat Rejection)

2. **Add the following paragraph to Section 1.6.2.12. See markup for location.**

   "Long-term core and suppression pool cooling can be performed by use of the LPCI with Heat Rejection mode for liquid breaks inside primary containment of sufficient size to support continuous recirculation as described below including the design basis LOCA. In the LPCI with Heat Rejection Mode, the RHRS pumps take suction from the suppression pool, and pump the water through the RHR heat exchanger where cooling takes place by transferring heat to the station cooling water systems. The fluid is then discharged back to the reactor vessel where sensible and decay heat is absorbed. The fluid returns to the suppression pool by flowing out the pipe break, into the drywell, and back to the suppression pool through the drywell to wetwell vent system. This method of cooling sets up a recirculation loop including the suppression pool, RHR heat exchanger, and reactor vessel."

3. **Change Section 1.6.2.12. See markup for location.**

   Insert "for a design basis LOCA" per the attached markup.

4. **Section 4.8.4 RHR Power Generation Design Basis**

   Remove item 2. "Provide suppression pool cooling for RCICS operation (SPC mode). This requirement is no longer a design requirement. Technical Specifications require depressurization at 120°F suppression pool temperature not 130°F [see markup]

4a. **Section 4.8.5.2 Shutdown Cooling Subsystem**

   Insert "torus return line" per the attached markup.

5. **Section 4.8.5.4 Suppression Pool Cooling Subsystem.**

   Remove statements evaluating RHR power generation design basis regarding RCICS operation." Some plants including PNPS had a RCIC design requirement to maintain suppression pool temperature less than 130°F for a few hours so the plant could stay pressurized and at temperature using recirculation pumps waiting to return to power operation. Current Technical Specifications are more conservative with respect to pool temperature and include the requirement to scram at 110°F and then depressurization at 120°F suppression pool temperature not 130°F.

Therefore, this design requirement is no longer applicable since this condition is precluded by current Technical Specifications [Ref. 22] [see markup for deleted requirement].

6. **Add the following new subsection to Section 4.8.5.4**

"4.8.5.4.1 LPCI with Heat Rejection

Long-term core and suppression pool cooling can be performed by use of the LPCI with Heat Rejection mode for liquid breaks inside primary containment of sufficient size to support continuous recirculation as described below including the design basis LOCA. In the LPCI with Heat Rejection Mode, the RHRS pumps take suction from the suppression pool, and pump the water through the RHR heat exchanger where cooling takes place by transferring heat to the station cooling water systems. The fluid is then discharged back to the reactor vessel where sensible and decay heat is absorbed. The fluid returns to the suppression pool by flowing out the pipe break, into the drywell, and back to the suppression pool through the drywell to wetwell vent system. This method of cooling sets up a recirculation loop including the suppression pool, RHR heat exchanger, and reactor vessel.

In LPCI with Heat Rejection mode, with two RHR pumps in operation, the heat exchanger bypass valve remains in its full open normal position. No disruption of LPCI flow is required to enter this mode of suppression pool cooling. This configuration will provide maximum core cooling but does not provide rated heat removal because more than half of the two pump LPCI flow rate goes through the heat exchanger bypass line and not the heat exchanger.

Rated heat removal from the containment is obtained using LPCI with Heat Rejection mode by removal of one RHR pump from LPCI service, closure of the RHR heat exchanger bypass valve, and valve adjustments as necessary to establish 5100 gpm through the RHR heat exchanger."

7. **Replace Figure 4.8-3 "Residual Heat Removal System Heat Transfer Capability" with attached curve.**

The new curve adds 75°F capability curve for RHR heat exchanger on Figure 4.8-3.

8. **Revise Table 4.8-1**

Add the following items under "Heat Transfer Capability"[see markup]

Shutdown Cooling Mode (two heat exchangers)	44.1×10^6 BTU/Hr at 125°F RHR Inlet and 75°F Service Water
Containment Cooling Mode (LPCI with Heat Rejection, each heat exchanger)	66.1×10^6 BTU/Hr at 178°F RHR Inlet and 75°F Service Water

9. **Revise 5.2.4.10 Primary Containment Steam Quenching**

Replace all information after paragraph one with the following:

Technical specification limits on bulk suppression pool temperature ensure effective steam condensation during a LOCA or SRV discharge. The water temperature in the vicinity of the SRV discharge to the suppression pool is referred to in design and licensing documents as the local pool temperature. During plant transients and accidents involving SRV actuation, the suppression pool temperature is elevated by steam discharged from the primary system to the suppression pool through the relief valve discharge line. Past experimental data and plant experience indicated the potential for unstable steam condensation during safety-relief valve (SRV) operation at elevated pool temperatures (Reference 2) related to high water temperature near the SRV discharge. The unstable condensation process in a heated pool was observed to cause cyclical pressure loads on the pool boundary and structures inside the wetwell.

Further experimental data demonstrated that quenchers on the SRV discharge effectively eliminated the previously observed condensation oscillations at higher pool temperatures. The NRC issued an SER in August 1994 accepting the conclusions contained in NEDO-30832-A (Reference 4) regarding "T" quencher performance to mitigate steam condensation loads. On the basis of the experimental evidence provided to the NRC by the BWROG, the local pool temperature limits related to SRV discharges previously required by Reference 1 are eliminated.

Current Technical Specification limits on suppression pool temperature ensure bulk pool temperature remains within an acceptable range to condense steam discharged to the suppression pool during a LOCA or SRV actuation.

10. **Change Section 5.2.9 as follows:**

Replace reference 2 with:

2. U.S. Nuclear Regulatory Commission, "Suppression Pool Temperature Limits for BWR Containments," "USNRC Report NUREG-0783," November 1981.

Replace reference 4 with:

4. General Electric Company, "Elimination of Limit on BWR Suppression Pool Temperature for SRV Discharge with Quenchers," NEDO-30832-A, May 1995.

Replace reference 7 with:

7. General Electric Company, "Impact on Containment Pressure Temperature Response of Proposed Capping of Certain Drywell Spray Sparger Nozzles," EAS52-0587, Rev.1, May 1987.

Replace reference 8 with:

8. General Electric Company, "Information on the Effect of Reduced Drywell Spray Flow Rates,", EAS56-0888, September 1988.

10a. Add note "g" in Table 8.5-1 for both diesel generators [see attached markup]. The changes on the markup of Table 8.5-1 are coordinated with Change Request 2183 which is in-process as of 3/25/96 to change the table to reflect Calculation PS79, Rev. 4.

11. Replace the first paragraph of Section 10.5.5.3 Accident and Transient Operations with the following:

"Either RBCCW loop has sufficient capacity with two pumps operating to transfer the RHR System heat load plus an additional 1×10^6 Btu/hr heat load from other essential equipment during a design basis LOCA assuming a 65°F or 75°F service water inlet temperature. Table 10.5-2 provides RBCCW heat exchanger peak capacity, flow rates, and fluid temperatures for the design basis accident for both a 65°F and 75°F service water inlet temperatures."

12. Add the following information to Table 10.5-2 under the heading "<u>RBCCW Heat Exchangers</u>"

RBCCW heat transfer rate at peak suppression pool temperature, BTU/hr	66.1×10^6 (at 75°F SSW inlet temperature and 178°F suppression pool temperature)
	65×10^6 (at 65°F SSW inlet temperature and 165°F suppression pool temperature)
Reactor Building Closed Cooling Water flow rate, gpm	3200
Salt Service Water flow rate, gpm	4,500

13. Replace the beginning paragraphs of FSAR section 14.5.3.1.1 through bullet number 1 with the following:

The following assumptions and initial conditions were used in the calculation of the effects of a LOCA on the primary containment. The plant response to

the accident can be separated into two distinct phases: the short term response, and the long-term recirculation phase. The short term response includes that period of time in the accident up to 600 seconds when initiation of containment cooling is assumed. The peak drywell and wetwell airspace temperatures occur in this period of time and are not influenced by the performance of containment cooling. The long-term recirculation phase of the accident response is defined to begin at 600 seconds with the initiation of containment cooling and continue past the peak suppression pool temperature to the point of minimum NPSH margin.

Historically, the primary containment response has been established using the design value of 65°F for the SSW inlet temperature to the RBCCW heat exchanger. The following discussion of containment response includes analysis performed for the DBA LOCA using a site maximum SSW injection temperature of 75°F. In the following discussion the analysis that used a 75°F SSW injection temperature is referred to as the 75°F SSW Case, likewise the analysis based on a 65°F SSW injection temperature is referred to as the 65°F SSW Case.

1. The reactor is operating at full power with all valves in the Recirculation system open. Initial power for the 75°F SSW Case was increased to 102% consistent with the current standard based on the requirements of Regulatory Guide 1.49.

<u>65°F SSW Case</u>	<u>75°F SSW Case</u>
1998 MWt	2038 MWt

14. Replace bullet number 5 in Section 14.5.3.1.1 with the following:

5. For both the short and long-term analysis in the 65°F SSW Case, the feedwater flow was assumed to stop instantaneously at time zero. This conservatism is used because the relatively cold feedwater flow, if considered to continue, tends to depressurize the reactor vessel, thereby reducing the discharge of steam and water into the primary containment.

   Short-term containment response in the 75°F SSW Case is consistent with the 65°F SSW Case. For the 75°F SSW Case long-term analysis, feedwater flow into the RPV continues until the high-energy feedwater (above feedwater enthalpy of 201 BTU/lbm) is injected into the reactor vessel. This assumption is conservative for the long-term suppression pool temperature analysis because additional energy is added to the reactor vessel and containment.

15. Add the following to existing assumption number 6 as a new paragraph [see markup for location]:

For the 75°F SSW Case, the vessel depressurization rates were calculated using the Homogeneous Equilibrium critical flow model described in NEDO-21052, "Maximum Discharge of Liquid-Vapor Mixtures from Vessels." (Reference 6).

16. Insert heading "<u>65°F SSW Case</u>" above existing first paragraph of existing text in FSAR 14.5.3.1.2 Containment Response [see markup]

17. Insert the following new subsection at the end of the existing section 14.5.3.1.2 [see markup for location]

<u>75°F SSW Case</u>

For the 75°F SSW Case, the calculated pressure and temperature responses of the containment are shown on Figures 14.5-16, and 14.5-17. The short-term response of the drywell, wetwell, and suppression pool is the same as for the Case B from the 65°F SSW Case. The containment response prior to 600 seconds is unaffected by containment cooling and remains the same for both cases. The 65°F SSW Case provides additional description of the short-term response.

Prior to the activation of containment cooling, the LPCI and Core Spray pumps have been adding liquid to the reactor vessel. After the vessel is flooded to the height of the jet pump nozzles, the excess flow discharges through the break into the drywell. This flow cools the fuel and flushes sensible heat from the reactor vessel into the drywell. The flow of subcooled liquid into the drywell causes a depressurization of the containment as the steam in the drywell is condensed.

For the 75°F SSW Case, the long-term analysis assumes one RHR loop is available for containment cooling. At 600 seconds, the necessary valves are opened admitting cooling water flow to the RHR heat exchanger. The RHR heat exchanger bypass valve is assumed to remain in its full open normal position and the RHR system is assumed to remain in LPCI mode with containment cooling by heat rejection through the RHR heat exchanger. No disruption of LPCI flow is required to enter this mode of cooling. This configuration will provide maximum core cooling but does not provide rated heat removal because more than half of the two pump LPCI flow rate goes through the heat exchanger bypass line and not the heat exchanger.

At 2 hours after the start of the accident a transition is made from two pump LPCI-Heat Rejection Mode to a one pump LPCI-Heat Rejec[illegible] Mode to provide rated heat removal from the containment. This transitio[illegible] requires removal of one RHR pump from LPCI service, closure of the RHR [illegible] exchanger bypass valve, and valve adjustments as necessary to establish 5100

gpm through the RHR heat exchanger. One pump LPCI-Heat Rejection Mode is assumed to run continuously throughout the remainder of the accident response.

The peak suppression pool temperature of 178°F, which is much less than the containment temperature design limit of 281°F occurs between 5 to 6 hours after the start of the accident. The suppression pool temperature response is shown on Figure 14.5-17 and a summary of equipment capability assumed in the analysis is shown in Table 14.5-2.

18. Add new Table 14.5-2 [see attached]

19. Add new Figures 14.5-16, 14.5-17, 14.5-18, 14.5-19 [see attached]

20. Replace Section 14.5.3.1.3 "Core Standby Cooling System Pump Net Positive Suction Head" with the following [see markup]

Note: The changes to this section by SE2971 should be made before the following changes. These changes have been coordinated with the new 65°F NPSH figures included in SE2971. Those figures numbered 14.5-9, 14.5-10, and 14.5-13 are referenced in the following discussion but are not included in this safety evaluation.

14.5.3.1.3 Core Standby Cooling System Pump Net Positive Suction Head

To assure proper operation of the CSCS pumps following a design basis LOCA, the primary containment and CSCS pump system design is such that Net Positive Suction Head (NPSH) margin is available to the pumps at all times.

The NPSH available (NPSHA) at the suction to the CSCS pumps is equal to the total absolute pressure minus the vapor pressure of water at the suppression pool temperature. The NPSH required at the pump suction (NPSHR) is the minimum pressure over and above the vapor pressure that must be present in order to prevent pump cavitation.

NPSH design margin is based on calculations that include the effect from the increase in wetwell vapor pressure and air/nitrogen partial pressure in equilibrium with increasing suppression pool temperature with an accounting for containment initial conditions and leakage.

The design margin for NPSH available to the RHR and Core Spray pumps is determined using the following assumptions:

1. The primary containment is assumed to contain the minimum credible mass of noncondensible gas (air/nitrogen) prior to the design basis LOCA. The drywell initial condition is 150°F, 80% RH, 1.3 psig, and the wetwell is 80°F, 100% RH, 0 psig.

2. The water vapor pressure in containment increases to be in equilibrium with the suppression pool temperature.

3. The partial pressure of the containment air/nitrogen increases with the pool temperature per the ideal gas laws after the initial mixing of the drywell and wetwell air has occurred.

4. Where stated on the figures, containment leakage has been calculated based on a leak rate of 1% per day for design basis conditions and 5% per day to demonstrate conservative design margin with impaired containment integrity. The leakage values represent percent mass per day at a reference pressure of 45 psig using the mass leakage formulation described in Appendix R.5.4.2 "Long Term Containment Response".

5. The suppression pool temperature profile is based on minimum primary containment system cooling, i.e., one RHR loop in containment cooling is assumed, with an initial suppression pool temperature of 80°F and either a Salt Service Water heat sink temperature of 65°F or a Salt Service Water heat sink temperature of 75°F.

6. Minimum initial water volume in the suppression pool is assumed (84,000 ft^3).

7. Drywell free volume temperature is equal to wetwell temperature following the accident. This is based on the redistribution of noncondensible gases between the drywell and wetwell via the vacuum-breaker system following the vessel depressurization phase.

8. Maximum required flow rates are used for the CSCS pumps to maximize the suction line losses and NPSH required by the pumps. The NPSHR is 23 ft at 5100 GPM for the RHR pumps and 29 ft at 4400 GPM for the Core Spray pumps.

Based on the above conservative assumptions, the margin for NPSH available was evaluated for the limiting accident event which is the design basis LOCA. The NPSH available and NPSH margin for the RHR and Core Spray pumps were evaluated for both a 75°F SSW injection temperature and a 65°F SSW injection temperature. In the following discussion the analysis that used a 75°F SSW injection temperature is referred to as the 75°F SSW Case, likewise the analysis based on a 65°F SSW injection temperature is referred to as the 65°F SSW Case.

Figure 14.5-9 shows the NPSH available as a function of pool temperature with zero containment leakage which makes this curve independent of time. Since no leakage effect is included, Figure 14.5-9 represents the highest

NPSH margin that can be obtained using the above assumptions and as can be seen, a large margin exists for all pool temperatures. NPSH margin for the 65°F SSW Case , with leakage effects included, is presented in a different format on Figure 14.5-10 and 13. Here, the suppression pool temperature and containment pressure are shown as a function of time. Also shown is the primary containment pressure required to provide the required NPSH to the RHR and Core Spray pumps at their maximum required flow rates. As can be seen, substantial margin exists throughout the duration of the event. Therefore, it can be concluded that adequate NPSH will be available at all times following a design basis LOCA for the 65°F SSW Case.

NPSH margin for the 75°F SSW Case , with leakage effects included, is presented on Figure 14.5-18 and 19. Here, the suppression pool temperature and containment pressure are shown as a function of time. Also shown is the primary containment pressure required to provide the required NPSH to the RHR and Core Spray pumps at their maximum required flow rates. As can be seen, substantial margin exists throughout the duration of the event. Therefore, it can be concluded that adequate NPSH will be available at all times following a design basis LOCA for the 75°F SSW Case.

The RHR and Core Spray system design analysis shows positive NPSH margin is available following the bounding design basis LOCA without containment positive pressurization beyond the initial pressurization that can be obtained from the nitrogen gas stored in the drywell and wetwell airspace. The design margin for NPSH available is that which exists between the minimum containment pressure that provides the required NPSH and the containment pressure that exists due to equilibrium conditions for the gas/vapor mixture with an accounting for containment initial conditions and leakage.

[continue with existing FSAR in this subsection beginning with the following]

During the Reactor Core Isolation Cooling System (RCICS) operation, ...(continue)

21. **Add the following as reference 6 to Reference list in Section 14.7 [see attached markup]**

6. General Electric Company, "Maximum Discharge Rate of Liquid-Vapor Mixtures from Vessels", Moody F.J., NEDO-21052, September 1975.

3. Core Spray System

The Core Spray System consists of two independent pump loops that deliver cooling water to spray spargers over the core. The system is actuated by conditions indicating that a breach exists in the nuclear system process barrier, but water is delivered to the core only after reactor vessel pressure is reduced. This system provides the capability to cool the fuel by spraying water onto the core.

4. Low Pressure Coolant Injection

LPCI is an operating mode of the RHRS but is discussed here because the LPCI mode acts as an engineered safeguard in conjunction with the other Standby Cooling Systems. LPCI uses the pump loops of the RHRS to inject cooling water at low pressure into an undamaged reactor recirculation loop. LPCI is actuated by conditions indicating a breach in the nuclear system process barrier, but water is delivered to the core only after reactor vessel pressure is reduced. LPCI operation, together with the core shroud and jet pump arrangment, provides the capability of core reflooding following a LOCA in time to prevent fuel clad melting.

Change title see item 1

1.6.2.12 Residual Heat Removal System (Containment Spray, ~~and~~ Suppression Pool Cooling,)

The suppression pool (torus) cooling subsystem is placed in operation to limit the temperature of the water in the suppression pool following a design basis LOCA. In the suppression pool cooling mode of operation the RHRS main system pumps take suction from the suppression pool, and pump the water through the RHRS heat exchangers where cooling takes place by transferring heat to the station cooling systems. The fluid is then discharged back to the suppression pool.

insert new paragraph item 2.

Another portion of the RHRS is provided to spray water into the containment as an augmented means of removing energy from the containment following a LOCA. This capability in excess of the required energy removal capability for a design basis LOCA and can be placed into service at the discretion of the operator.

1.6.2.13 Control Rod Velocity Limiter

A control rod velocity limiter is attached to each control rod to limit the velocity at which a control rod can fall out of the core should it become detached from its control rod drive. The rate of reactivity insertion resulting from a rod drop accident is limited by this action. The limiters contain no moving parts.

1.6.2.14 Control Rod Drive Housing Supports

Control rod drive housing supports are located underneath the reactor vessel near the control rod housings. The supports limit the travel of a control rod in the event that a control rod housing is ruptured.

4.8 RESIDUAL HEAT REMOVAL SYSTEM

4.8.1 Safety Objective

The safety objective of the Residual Heat Removal System (RHRS) is to provide core cooling, in conjunction with other Core Standby Cooling Systems (CSCS), and to provide containment cooling as required during abnormal operational transients and postulated accidents.

4.8.2 Safety Design Basis

The safety design basis of the RHRS is to:

1. Restore and maintain the coolant inventory in the reactor vessel after a loss of coolant accident (LOCA) as required for core cooling in conjunction with other CSCS (low pressure coolant injection (LPCI) mode)
2. Provide cooling for the suppression pool and thereby remove heat from the containment following a LOCA to reduce containment pressure (suppression pool cooling (SPC) mode)
3. Maintain pressure suppression pool temperature during normal operation to within the limits assumed in the Station Safety Analysis
4. The suppression pool shall be the source of water for the LPCI mode of operation of the RHRS in order to provide a complete recycle path for water lost from the reactor vessel following reflooding
5. To provide a high degree of assurance that the RHRS operates satisfactorily during a LOCA, each active component shall be capable of being tested during operation of the nuclear system

4.8.3 Power Generation Objective

The power generation objective of the RHRS is to provide residual heat removal (RHR) capability when the main condenser heat sink is unavailable.

4.8.4 Power Generation Design Basis

The power generation design basis of the RHRS is to:

1. Remove residual heat from the nuclear system so that refueling and nuclear system servicing can be performed (shutdown cooling (SDC) mode)

delete

2. Provide suppression pool cooling for RCICS operation (SPC mode)

2 ~~3~~. Supplement the Fuel Pool Cooling System capacity when necessary to provide additional cooling capacity

RHR equipment is designed in accordance with Class I seismic design criteria (see Section 12 and Appendix C). The system is assumed to be filled with water for the seismic analysis.

The system piping and main system pumps are designed, constructed, and tested in accordance with the requirements of Appendix A. The pumps are also designed and constructed in accordance with the Standards of the Hydraulic Institute. The shell side of the heat exchangers are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class C vessels, and the tube side is designed in accordance with Section VIII. The provisions of the Winter Addenda of 1966, paragraph N2113 apply.

Power for four RHR pumps (two pumps on each of two loops) is normally provided through the two independent 4,160V emergency buses. In the event that the normal auxiliary power supply is not available, the 4,160V buses serving the two loops of RHR are powered separately from the two diesel generators.

Additional flexibility has been provided in the design by the addition of a permanent piping connection from the station salt water service pumps to the RHR Piping System. This connection is sized to provide 5,000 gal/min at 0 psig reactor pressure. All piping and equipment in the Service Water System which serves Class I equipment and this additional piping connection are designed to Class I seismic requirements.

The interconnection of the service water and RHR is manually initiated. Inadvertent admission of salt water to the RHR is prevented by requiring the operator to switch over a spectacle flange, and to open two locked-closed valves. Leaks from either system can be detected by periodic inspections of locked-open drains on each side of the spectacle flange.

4.8.5.2 Shutdown Cooling Subsystem

The Shutdown Cooling (SDC) Subsystem is an integral part of the RHR and is placed in operation during a normal shutdown and cooldown. The initial phase of nuclear system cooldown is accomplished by dumping steam from the reactor vessel to the main condenser with the main condenser acting as the heat sink. When nuclear system pressure has decreased to 50 psig the steam supply pressure is no longer sufficient to maintain vacuum in the main condenser, and the RHR is placed in the SDC mode of operation to complete cooldown of the nuclear system. The SDC Subsystem is capable of reducing reactor water temperatures to 125°F approximately 20 hr after reactor shutdown so that the reactor can be refueled and serviced. In the SDC mode, reactor coolant is pumped by the RHR main system pumps from one recirculation loop through the RHR heat exchanger(s) where heat is transferred to the Reactor Building Closed Cooling Water System (RBCCW). Reactor coolant is then returned to the reactor vessel through connections to the recirculation loop(s). Shutdown cooling suction valves are interlocked with the torus suction and torus return line valves valves so that they cannot be

region through one of the recirculation loops. Instrumentation is provided to detect the undamaged path for injection of LPCI flow (see Section 7.4, Core Standby Cooling Systems Control and Instrumentation). Water lost from the vessel through a break in the piping within the primary containment returns to the suppression pool through the pressure suppression vent pipes. It is concluded that safety design basis 4 is satisfied.

Coolant flow to the RHR heat exchangers from the Reactor Building Closed Cooling Water System is not required immediately after a LOCA because heat rejection from the containment is not necessary during the time it takes to flood the reactor.

4.8.5.4 Suppression Pool Cooling Subsystem

The Suppression Pool Cooling (SPC) Subsystem is an integral part of the RHR and is placed in operation to remove heat from the pressure suppression pool to reduce pressure in the primary containment following a LOCA. This system is also operated as required during planned operations to control suppression pool water temperatures within the limits assumed in the Station Safety Analysis.

With the RHR in the SPC mode of operation, the RHR pumps are aligned to pump water from the suppression pool through the RHR heat exchangers, where cooling takes place by transferring heat to the Reactor Building Closed Cooling Water System. Torus cooling return line valves are interlocked with the shutdown cooling suction valves, such that they cannot be opened simultaneously. The flow returns to the suppression pool via return lines which discharge below to the pool surface.

The RHR in the SPC mode functions to transfer heat from the primary containment to the Reactor Building Closed Cooling Water System thereby lowering the primary containment pressure. It is concluded safety design bases 2 and 3 are satisfied by this mode of RHR operation. In the event of reactor vessel isolation, the RHR in the SPC mode is capable of maintaining the torus water temperature below 130°F for at least 2 hr of RCIC operation. It is concluded that power generation design basis 2 is satisfied by this mode of RHR operation

delete see item 5.

Insert new subsection 4.8.5.4.1 see item 6.

4.8.5.5 Containment Spray Subsystem

The Containment Spray Subsystem provides containment spray capability as an alternate method for reducing containment pressure following a LOCA. A portion of the water pumped through the RHR heat exchangers can be diverted to spray headers in the drywell and above the suppression pool. The portion of the RHR heat exchanger flow to the spray headers in the drywell condense any steam that may exist in the drywell thereby lowering containment pressure. The remaining portion returns to the suppression pool via the suppression pool bypass line. The spray collects in the bottom of the drywell until the water level rises to the level of the pressure suppression vent pipes where it overflows, and drains back to the suppression pool. Approximately 5 percent of the

TABLE 4.8-1

RESIDUAL HEAT REMOVAL SYSTEM EQUIPMENT DESIGN DATA

Number Installed - 4	Design Temperature - 350°F
Shutoff Head - 600 ft	Design Pressure - 450 psi

Three Pump Injection at 20 PSIG Reactor Vessel Pressure (LPCI mode)

Capacity (each)	4,800 gal/min
Total Dynamic Head	420 ft

Two Pump Injection at 20 PSIG Reactor Vessel Pressure (LPCI mode)

Capacity (each)	5,500 gal/min
Total Dynamic Head	220 ft

HEAT EXCHANGERS

Number Installed - 2
Shell Side Fluid - Reactor Water or Suppression Pool Water
Tube Side Fluid - Reactor Building Closed Cooling Water
Design Pressure - 450 psig Design Temperature 40-350°F
Pressure Drop at Design Conditions - shell and tube side - 10 psi

Heat Transfer Capability

Shutdown Cooling Mode (two heat exchangers)	78×10^6 BTU/Hr at 125°F RHR Inlet and 55°F Service Water
Containment Cooling Mode (each heat exchanger)	64×10^6 BTU/Hr at 166°F RHR Inlet and 65°F Service Water

Insert 75°F data for shutdown Cooling Mode
see item 8.

Insert 75°F data for Containment
Cooling with LPCI with Heat Rejection
See item 8.

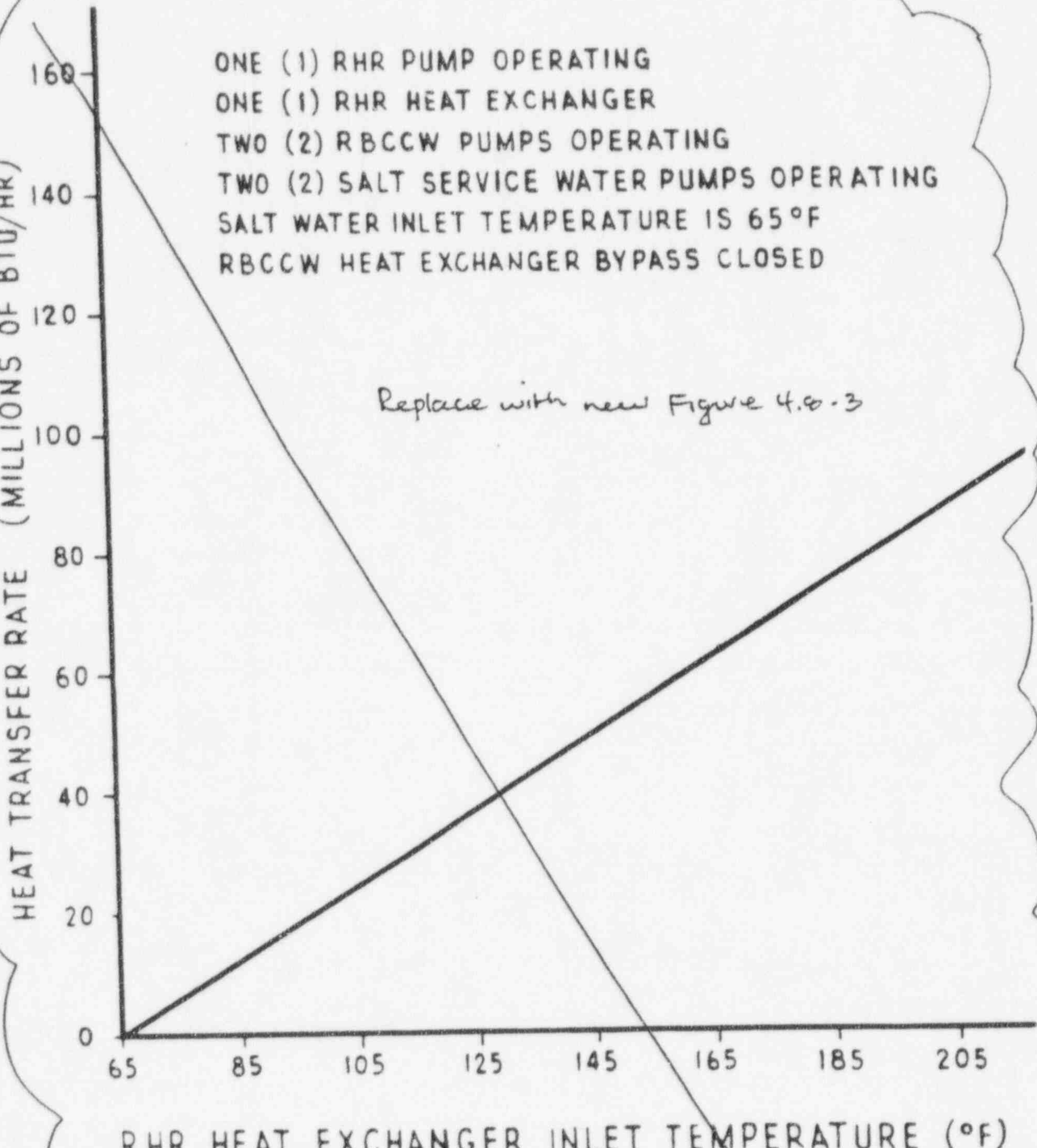

FIGURE 4.8-3
RESIDUAL HEAT REMOVAL SYSTEM
HEAT TRANSFER CAPABILITY
PILGRIM NUCLEAR POWER STATION
FINAL SAFETY ANALYSIS REPORT

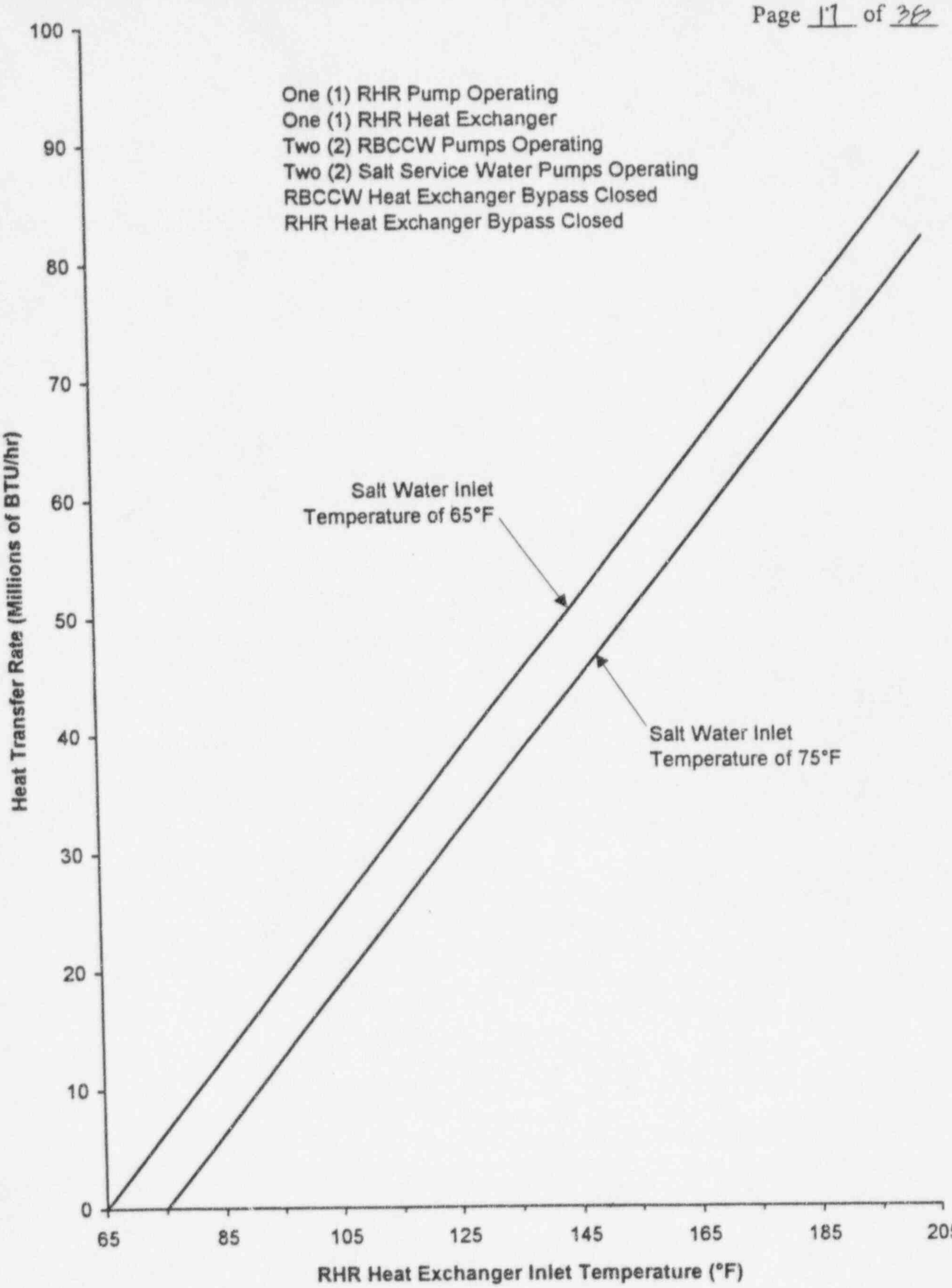

Figure 4.8-3

Residual Heat Removal System Heat Transfer Capability

Pilgrim Nuclear Power Station

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Because the amount of radioactive materials in the reactor coolant is small, a sufficient limitation of fission product release will be accomplished if the isolation valves are closed before the coolant drops below the top of the core.

It is concluded that safety design basis 6 is met.

5.2.4.7 Containment Flooding

As is discussed in Section 12, the PCS is designed for the conditions associated with flooding the containment.

It is concluded that safety design basis 4 is met.

5.2.4.8 Pressure Suppression Pool Water Storage

Based upon the Station Safety Analysis presented in Section 14, the quantity of water stored in the suppression pool is sufficient to condense the steam from a DBA and to provide water for the CSCS. As discussed in Section 12, the suppression pool is considered in the loading conditions on the PCS.

It is concluded that safety design basis 7 is met.

5.2.4.9 Limitations During Planned Operations

As is discussed in Sections 5.2.3.6, 5.2.3.7, and 5.2.3.8, the PCS is designed to be kept within the limits of parameters assumed in the Station Safety Analysis presented in Section 14 during planned operations.

It is concluded that safety design basis 8 is met.

5.2.4.10 Primary Containment Steam Quenching

The suppression chamber, or torus, is designed to contain a pool of water in order to suppress the pressure during a postulated LOCA by condensing the steam released from the Reactor Primary System. The reactor system energy released by relief valve operation during operating transients also is released into the suppression pool.

Replace with new text see item 9.

As a result of concerns regarding potential instability of steam condensation in a hot suppression pool, the United States Nuclear Regulatory Commission (NRC) has imposed pool temperature limits for plant transients involving safety/relief valve (SRV) operation (Reference 1). The limits which ensure smooth steam condensation for discharge through quenchers are:

1. For all plant transients involving SRV operation during which the steam flux through the quencher perforations exceeds 94 lbm/ft^2-sec, the suppression pool local temperature shall not exceed 200°F.

2. For all plant transients during which the steam flux through the quencher perforations is less than 42 lbm/ft^2-sec, the suppression pool local temperature should be at least 20°F subcooled. This corresponds to a local temperature limit of 201.4°F for PNPS.

3. For plant transients involving SRV operation during which the steam flux through the quencher perforations exceeds 42 lbm/ft^2-sec but is less than 94 lbm/ft^2-sec, the suppression pool local temperature can be established by linearly interpolating the local temperatures established under items (1) and (2) above.

These limits are depicted in Figure 5.2-19.

An analysis was done (Reference 2) to show that PNPS complies with the NRC criteria.

Seven transient events have been identified, one of which is expected to result in the maximum long-term suppression pool temperature. The seven events are as follows:

1A. Stuck-open SRV during power operation with one RHR loop available.

1B. Stuck-open SRV during power operation assuming reactor isolation due to MSIV closure.

2A. Isolation/scram and manual depressurization with one RHR loop available.

2B. Isolation/scram and manual depressurization with the failure of an SRV to reclose (SORV).

2C. Isolation/scram and manual depressurization with two RHR loops available. This case demonstrates the pool temperature responses when an isolation/scram event occurs under normal power operation (i.e., when all systems are operating in normal mode).

3A. Small-break accident (SBA) with manual depressurization: accident mode with one RHR loop available.

3B. Small-break accident (SBA) with manual depressurization and failure of the shutdown cooling system.

The analysis indicated that the maximum temperature occurs during Case 2A. The maximum local pool temperature for this case is 199°F (reference 4) which is less than the 201.4°F limit applicable for low steam flux conditions.

Replace with new text, see item 9.

therefore be concluded that the resultant radiological exposures for the above pipe failures will at the maximum be based on only that activity contained in the primary coolant, which is discharged to the secondary containment.

To provide an upper limit to the radiological exposures, the assumptions have been made that:

1. All of the primary coolant which contained activity is eventually discharged to the secondary containment
2. Considering the thermodynamics of the coolant discharged, a maximum of 1/3 of the coolant is flashed to steam resulting in the release to the secondary containment of 1/3 of the coolant activity
3. Consideration of the condensing and plateout surfaces that the released steam will have to come in contact with prior to being released from the top of the Reactor Building results in a minimum reduction factor of 3 for the released iodine activity
4. The activity is released from the top of the Reactor Building under those meteorological conditions, which maximize the offsite exposures
5. The activity contained in the reactor coolant is consistent with an offgas emission rate of 10^5 microcuries/sec

Based on the above considerations, the resultant site boundary thyroid dose is 0.08 rem while the LPZ thyroid dose is 0.002 rem. If the conservative assumption is made that downwash of the released effluent occurs and that the coolant activity is at a level consistent with the technical specification offgas activity (i.e., 0.9 ci/sec), the resultant site boundary thyroid dose is 15 rem and the LPZ thyroid dose is 0.6 rem, both of which are well below the 300 rem guideline set forth in 10CFR100.

5.2.9 References

1. Bodega Bay Preliminary Hazards Report, Appendix I, Docket 50-205, December 28, 1962.

Replace with new Ref. 2 see item 10.

2. General Electric Company, "PNPS Unit 1 Suppression Pool Temperature Response," NEDC-22089-P, March 1982.
3. J. M. Caroll, BECo Letters to NRC, May 15, 1973.

- Replace with new Ref. 4 see item 10.

4. General Electric Company, "Pilgrim Suppression Pool Temperature Analysis," Letters from R. Thibault to G. McHugh, December 1982.
5. Franklin Research Center Technical Evaluation Report, "Containment Leakage Rate Testing", TER-C5257-40, May 5, 1981.

6. NRC letter, "Licensee Response to IE Bulletin 79-08 and Acceptability of Single Check Valves as Containment Isolation for Pilgrim," Ronald Eaton (NRC) to G. W. Davis, February 4, 1991.

7. E. H. Hoffman, et. al., General Electric Company, "Drywell Temperature Analysis for Pilgrim Nuclear Power Station," EAS-98-0887; August, 1987.

8. Letter, GE to BECo, "Safety Evaluation of Proposed Capping of Certain Drywell Spray Nozzles," G-HK-7-157, dated April 20, 1987.

Replace with new Reference 7 and 8, see item 10.

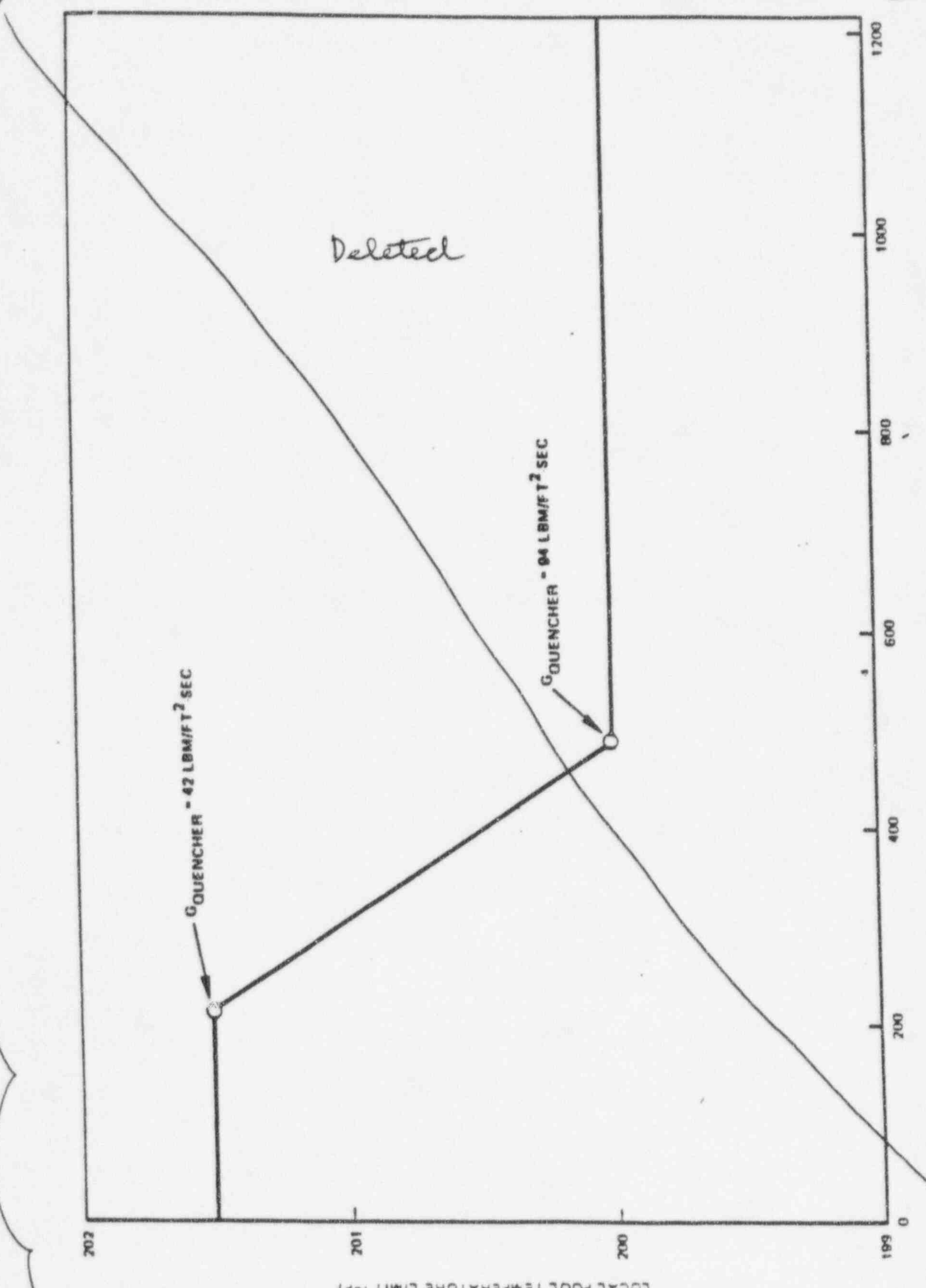

Figure 5.2-19. NRC Specified Local Pool Temperature Limit Based on NUREG-0783 for Pilgrim

Revision 5 - July 1985

-FSAR
TABLE 8.5-1

DIESEL GENERATOR "A" EMERGENCY LOADS (c) STANDBY AC POWER SYSTEM

Function	Station Total Number Available	Maximum Number Available	Nameplate HP	Load KW	LOCA w/ Loop 0-10 Min. Number Connected	LOCA w/ Loop 0-10 Min. Load KW	LOCA w/ Loop >10 Min. Number Connected	LOCA w/ Loop >10 Min. Load KW (a)	Loop Number Connected	Loop Load KW
Control Rod Drive Feed Pump	2	1	250	227	-	-	-	-	1	227
Residual Heat Removal Pump	4	2	800	639	2	1278	1	639	1	639
Core Spray Cooling Pump	2	1	800	639	1	639	1	639	-	-
Battery Charger (125VDC)	3	2	N/A	32	2 ~~1~~	64 ~~32~~(f)	2 ~~1~~	64 ~~32~~(f)	2 ~~1~~	64 ~~32~~(f)
Aux Oil Pump Recirc M-G Set	2	1	30	25	-	-	-	-	1	25
Standby Gas Treatment System	2	1	N/A	36	1	36	1	36	1	36
RBCCW Pump	6	3	60	54 ~~a~~ e	1	54	2	108	2	108
TBCCW Pump	2	1	100	81	1	81	1	81	1	81
Salt Service Water Pump	5	3	100	83 ~~a~~ e	1	83	2	166	2	166
Station Instrument Air Compr.[b]	3	2	40	33	-	-	2	67	2	67
Drywell Unit Coolers	16	8	87[d]	87	-	-	-	-	8	87
Emergency Lighting	-	All	N/A	36	All	36	All	36	All	36
~~Instrumentation & Control~~			~~N/A~~	~~79~~		~~79~~		~~79~~		~~79~~
Vital MG Set	-	1	75	15	1	15	1	15	1	15
Station HVAC System[b]	-	-	-	-	-	62 ~~38~~	-	118 ~~30~~	-	~~30~~ 118
Turbine Generator Auxiliaires	-	-	-	113	-	21	-	113	-	113
Miscellaneous Auxiliaries	-	-	-	-	-	129 ~~34~~	-	286 270	-	~~293~~ 277
345KV SWYD Feed	-	-	-	75	-		-		-	75
Total KW Load on KW. "A" Continuous	-	-	-	-	-	2562(g) ~~2463~~	-	2416 ~~2317~~	-	~~2099~~ 2198
Total Short Time (MOV's) KW Load on D.G. "A"	-	-	-	181[c]	-	181[c]	-	77[c]	-	77[c]
Total KW Load (Continous & Short Term)						2743 ~~2644~~		2493 ~~2394~~		~~2176~~ 2275
Battery charger (250VDC)	2	1	N/A	64	1	64(h)	1	64(h)	1	64(h)

Insert

NOTES:

a. Of these additional loads, approximately 250KW ~~225 KW~~ could ~~are~~ automatically restarted when the operator manually stops any one of the three large CSCS pumps. The other load additions in this column are started manually by the operator as required. when diesel load limits permit.

b. Intermittent or short term loads.

c. Isolation and essential valves.

d. Equivalent hp values.

e. Includes essential CSCS unit area coolers, battery room exhaust fan, and essential control room ventilation.

Add notes (f), (g), and (h)

f. 32 KW of this loading assumes an abonormal lineup where the backup 125VDC charger feeds the "A" battery.

g. After short term MOV's operate, with three CSCS pumps operating the total diesel generator load remains within the 2,000 hr rating (2750 KW) with manual start of one additional SSW pump and one addtional RBCCW pump to support containment cooling.

h. This loading assumes an abnormal lineup where the backup charger feeds the 250VDC battery.

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TABLE 8.5-1

DIESEL GENERATOR "B" EMERGENCY LOADS (c) - STANDBY AC POWER SYSTEM

	Station				LOCA w/ Loop				Loop	
					0-10 Min.		>10 Min.			
Function	Total Number Available	Maximum Number Available	Nameplate HP	Load KW	Number Connected	Load KW	Number Connected	Load (a) KW	Number Connected	Load KW
Control Rod Drive Feed Pump	2	1	250	227	-	-	-	-	1	227
Residual Heat Removal Pump	4	2	800	639	2	1278	1	639	1	639
Core Spray Cooling Pump	2	1	800	639	1	639	1	639	-	-
Battery Charger (125VDC)	3	2	N/A	32	2 *	64 ~~32~~ (f)	2 *	64 ~~32~~ (f)	2 *	64 ~~32~~ (f)
Battery Charger (250 VDC)	2	1	N/A	64	1	64	1	64	1	64
Standby Gas Treatment System	-	-	-	36	1 ×	36	1 ×	36	1 ×	36
RBCCW Pump	6	3	60	54 (e)	1	54	2	108	2	108
TBCCW Pump	2	1	100	81	1	81	1	81	1	81
Salt Service Water Pump	5	3	100	83 (e)	1	83	2	166	2	166
Aux Oil Pump Recirc M.G. Set	2	1	30	25	-	-	-	-	1	25
Station Instrument Air Compr.[b]	3	2	40	33	-	-	2	67	2 *	67
Emergency Lighting	-	-	N/A	36	All	36	All	36	All	36
Drywell Unit Coolers	16	8	87[d]	-	-	-	-	-	8	87
~~Instrumentation & Control~~	-	-	-	-		~~79~~		~~79~~	-	~~79~~
Vital MG Set	-	-	75	15	1	15	1	15	1 ×	15
Station HVAC System[b]	-	-	-	-	-	64 ~~30~~	-	120 ~~22~~	-	120 ~~22~~
Turbine Generator Auxiliaires	-	-	-	113	-	21	-	113	-	113
Miscellaneous Loads	-	-	-	-	-	128 ~~74~~	-	316 ~~332~~	-	322 ~~339~~
345KV SWYD Feed	-	-	-	-	-	-	-	-	-	75
Total KW Load on "B" Continuous	-	-	-	-	-	~~2529~~ 2563 (g)		~~2429~~ 2463	-	2245 ~~2211~~
Total Short Time (MOV's) KW Load on D.G. "B"	-	-	-	-	-	151[c]	-	12[c]	-	12[c]
Total KW Load (Continous & Short Term)						2714 ~~2680~~		2476 ~~2441~~	-	2257 ~~2223~~

NOTES:

a. Of these additional loads, 225 KW approximately 250 KW ~~are~~ could automatically restarted when the operator manually stops any one of the three large CSCS pumps. The other load additions in this column are started manually by the operator as required when diesel load limits permit.

b. Intermittent or short term loads.

c. Isolation and essential valves.

d. Equivalent hp values.

e. Includes essential CSCS unit area coolers, battery room exhaust fan, and essential control room ventilation.

Add notes (f) and (g)

f. 32 KW of this loading assumes an abonormal lineup where the backup 125VDC charger feeds the "B" battery.

g. After short term MOV's operate, with three CSCS pumps operating the total diesel generator load remains within the 2,000 hr rating (2750 KW) with manual start of one additional SSW pump and one addtional RBCCW pump to support containment cooling.

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Two RBCCW pumps in each loop are required for cooldown with the RHR System in the shutdown cooling mode. The design heat transfer 20 hr after reactor shutdown is 68 x 10^6 Btu/hr from the RHR System to the RBCCW System. Other cooling loads increase the RBCCW System total design heat transfer to approximately 80 x 10^6 Btu/hr for this mode of operation. During planned operations, the RBCCW System also functions as an intermediate barrier between nuclear system equipment and the Salt Service Water System. The pressure in the RBCCW loops is higher than the pressure in the service water system to prevent salt water contamination of the RBCCW System. Detectors are located in the system to continuously monitor radioactivity level. On detection of a high radiation level, an alarm will be set off automatically in the control room.

10.5.5.3 Accident and Transient Operations

Replace with new text see item 11.

Either RBCCW loop has sufficient capacity with two pumps operating to transfer the design RHR System heat load of 64 x 10^6 Btu/hr (at an RHR System inlet temperature of 165°F) during postulated transient or accident conditions. The heat exchanger in each RBCCW loop is capable of transferring the design RHR heat load plus an additional 1 x 10^6 Btu/hr heat load from other essential equipment assuming a Salt Service Water System flow rate of 4,500 gpm at a salt water inlet temperature of 65°F.

Following a postulated loss of coolant accident (LOCA) coincident with loss of the preferred (offsite) ac power source, the operating RBCCW pumps will trip. One RBCCW pump in each loop is automatically restarted on its respective diesel generator approximately 30 sec after ac power is restored to the emergency service bus. Sufficient pressure is normally generated by one pump in each loop to preclude automatic starting of additional RBCCW pumps. In the event of pump failure, the low pressure condition in the pump discharge header will result in automatic sequential start of other pumps in the loop until the low pressure condition is corrected. Approximately 10 min later, the main control room operator will manually initiate cooling to the RHR System by starting a second RBCCW pump in each operational loop, isolating the nonessential loads on each loop, opening either one of the two inlet valves supplying RBCCW cooling water to the RHR heat exchanger in each loop, and closing the RBCCW heat exchanger bypass valves and temperature control valves.

The same automatic RBCCW pump start sequence also takes place in the event that the preferred (offsite) ac power source is lost without concurrent LOCA conditions.

Additional flexibility of system operation has been provided through the capability of interconnection of the two loops through the crossties. Thus, the system could still function under a variety of degraded conditions.

TABLE 10.5-2

REACTOR BUILDING COOLING WATER SYSTEM
EQUIPMENT DATA
NO. OF INDEPENDENT LOOPS - 2

RBCCW Pumps

Quantity per loop	3
Type	Horizontal Centrifugal
Flow and Head	1,700 gal/min at 100 ft TDH
Material:	
Casing/impeller/shaft	Cast Steel/Bronze/Stainless Steel
Motor: Size	60 hp
Voltage/phase/cycle	440 V/3 phase/60 cycle
rpm	1,750

RBCCW Heat Exchangers

Quantity per loop	1
Type	Horizontal, Shell and Tube, TEMA type AGM
Heat Transfer Area	10,200 ft^2 (effective)
Shell Design:	
Pressure/Temperature	150 psig/200°F
Material	Carbon Steel
Flow Medium	Inhibited Demineralized Water
Tube Design:	
Pressure/Temperature	150 psig/200°F
Materials: Tube	90-10 copper nickel
Tube sheet	90-10 copper nickel
Tube Joint	Rolled and welded
Flow Medium	Sea Water

Add heat removal and flow data here, see item 12.

14.5.3 Loss of Coolant Accident

Break of a large recirculation pipe represents the limiting pipe break inside the containment. This event has been analyzed quantitatively in Section 6.5. The following is a discussion of the containment analysis and radiological consequences. Assumptions used in these analyses are given in Appendix R.6 and below.

14.5.3.1 Primary Containment Response

14.5.3.1.1 Initial Conditions and Assumptions

Replace with new text see item 13.

The following assumptions and initial conditions were used in the calculation of the effects of a LOCA on the primary containment. These assumptions are in addition to those specified for the LOCA described in Appendix R.3.3.1.

1. The reactor is operating at a full power of 1,998 MWt with all valves in the Recirculation System open. This maximizes the break area and results in the most severe primary containment pressure transient.

2. The reactor is assumed to go subcritical at the time of accident initiation due to void formation in the core region. Scram also occurs in less than 1 sec from receipt of the high drywell pressure signal, but the difference in shut down time between 0 and 1 sec is negligible.

3. The sensible heat released in cooling the fuel to 545°F (the normal primary system operating temperature) and the core decay heat were included in the reactor vessel depressurization calculation. The rate of energy release was calculated using a conservatively high heat transfer coefficient throughout the depressurization. Because of this assumed high energy release rate, the vessel is maintained at near rated pressure for almost 6 sec. The high vessel pressure increases the calculated flow rates out of the break; this is conservative for containment analysis

purposes. With the vessel fluid temperature remaining near 545°F, however, the release of sensible energy stored below 545°F is negligible during the first 6 sec. The later release of this sensible energy does not affect the peak drywell pressure. The small effect of this energy on the end of transient suppression pool temperature is included in the calculations.

4. The main stream line isolation valves were assumed to start closing at 0.5 sec after the accident, and the valves were assumed to be fully closed in the shortest possible time of 3 sec following closure initiation. Actually, the closures of the main steam line isolation valves are expected to be the result of reactor low-low water level, so these valves may not receive a signal to close for over 4 sec, and the closing time could be as high as 10 sec. By assuming rapid closure of these valves, the reactor vessel is maintained at a high pressure, which maximizes the discharge of high energy steam and water into the primary containment, which in turn maximizes the loading on the containment.

Replace with new text see item 14..

5. The feedwater flow was assumed to stop instantaneously at time zero. This conservation is used because the relatively cold feedwater flow, if considered to continue, tends to depressurize the reactor vessel, thereby reducing the discharge of steam and water into the primary containment.

6. The vessel depressurization flow rates were calculated using Moody's critical flow model[2] assuming "liquid only" outflow because this maximizes the energy release to the containment. "Liquid only" outflow means that all vapor formed in the vessel due to bulk flashing rises to the surface rather than being entrained in the exiting flow. Some entrainment of the vapor would occur and would significantly reduce the reactor vessel discharge flow rates. Moody's critical flow model, which assumes annular, isentropic flow, thermodynamic phase equilibrium, and maximized slip ratio, accurately predicts vessel outflows through small diameter orifices. However, actual flow rates through larger flow areas are less than the model indicates due to the effects of a near homogeneous two phase flow pattern and phase nonequilibrium. These effects are in addition to the reduction due to vapor entrainment discussed above.

Add additional paragraph to bullet 6. see item 15.

7. The pressure response of the containment is calculated assuming:

   a. Thermodynamic equilibrium in the drywell and suppression chamber. Because complete mixing is nearly achieved, the error introduced by assuming complete mixing is negligible and in the conservative direction.

slightly higher than the test results. The overprediction is believed to be due to a combination of:

No condensation assumed in calculated response.

Slight overprediction of reactor vessel discharge flow rates, and

Incomplete liquid carryover into the drywell vents.

Insert Heading see item 16.

As the chosen size of the vessel orifice increases, the vessel depressurization rate is over predicted and the overprediction of drywell pressure increases. This trend is illustrated on Figure 14.5-4, where calculated and measured drywell peak pressure are compared. In no case did the model underpredict the test data.

14.5.3.1.2 Containment Response

65°F SSW Case

The calculated pressure and temperature responses of the containment are shown on Figures 14.5-5, 14.5-6, and 14.5-7. Figure 4.5-5 shows that the calculated drywell peak pressure is 45 psig, which is well below the maximum allowable pressure of 62 psig. After the discharge of the primary coolant from the reactor vessel into the drywell, the temperature of the suppression chamber water approaches 130°F (Figure 14.5-7), and the primary containment pressure stabilizes at about 27 psig, as shown on Figure 14.5-5. Most of the noncondensible gases are forced into the suppression chamber during the vessel depressurization phase. However, the noncondensibles soon redistribute between the drywell and the suppression chamber via the Vacuum-Breaker System as the drywell pressure decreased due to steam condensation.

The Core Spray System removes decay heat and stored heat from the core, thereby controlling core heatup and limiting metal water reaction to less than 0.1 percent. The core spray water transports the core heat out of the reactor vessel through the broken recirculation line in the form of hot water. This hot water flows into the suppression chamber via the drywell to suppression chamber vent pipes. Steam flow is negligible. The energy transported to the suppression chamber water is then removed from the Primary Containment System by the Residual Heat Removal System (RHRS) heat exchangers.

Prior to activation of the RHRS containment cooling mode (arbitrarily assumed at 600 sec after the accident), the RHRS pumps (low pressure coolant injection (LPCI) mode) have been adding liquid to the reactor vessel along with the core spray. After the reactor vessel is flooded to the height of the jet pump nozzles, the excess flow discharges through the recirculation line break into the drywell. This flow, in addition to cooling the fuel, offers considerable cooling to the drywell and causes a depressurization of the containment as the steam in the drywell is condensed. At 500 sec,

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Table 14.5-2

LOSS OF COOLANT ACCIDENT PRIMARY CONTAINMENT RESPONSE SUMMARY

	75°F SSW Case
Peak Wetwell Pressure, psig	22.1
Peak Pool Temperature,°F	178
Mode of RHR Operation	2 Pump LPCI-Heat Rejection Mode and 1 Pump LPCI-Heat Rejection Mode
Number of RHR Loops operating	1
Number of RHR pumps operating	2 Pumps from 600 seconds to 7200 seconds 1 Pump from 7200 seconds to 30 days
Number of RHR heat exchangers	1
Total LPCI - Heat Rejection Mode flowrate, gpm	9,500 at zero reactor pressure (2 pump) 5,100 at zero reactor pressure (1 pump)
Core Spray System flow rate, gpm	4,100 gpm at zero reactor pressure
RHR heat exchanger flow rate, gpm	3,430 (2 pump LPCI-Heat Rejection Mode) 5,100 (1 pump LPCI-Heat Rejection Mode)
RHR heat transfer rate at peak suppression pool temperature, BTU/hr	66.1×10^6
Number of RBCCW loops operating	1
Number of RBCCW pumps operating	2 each, 10 minutes after the accident
Number of RBCCW heat exchangers operating	1
Total RBCCW flow rate to RHR, gpm	3200
Number of SSW loops operating	1
Number of SSW pumps operating	2 each, 10 minutes after the accident
Total SSW flow rate to RBCCW	4,500
SSW inlet water temperature,°F	75

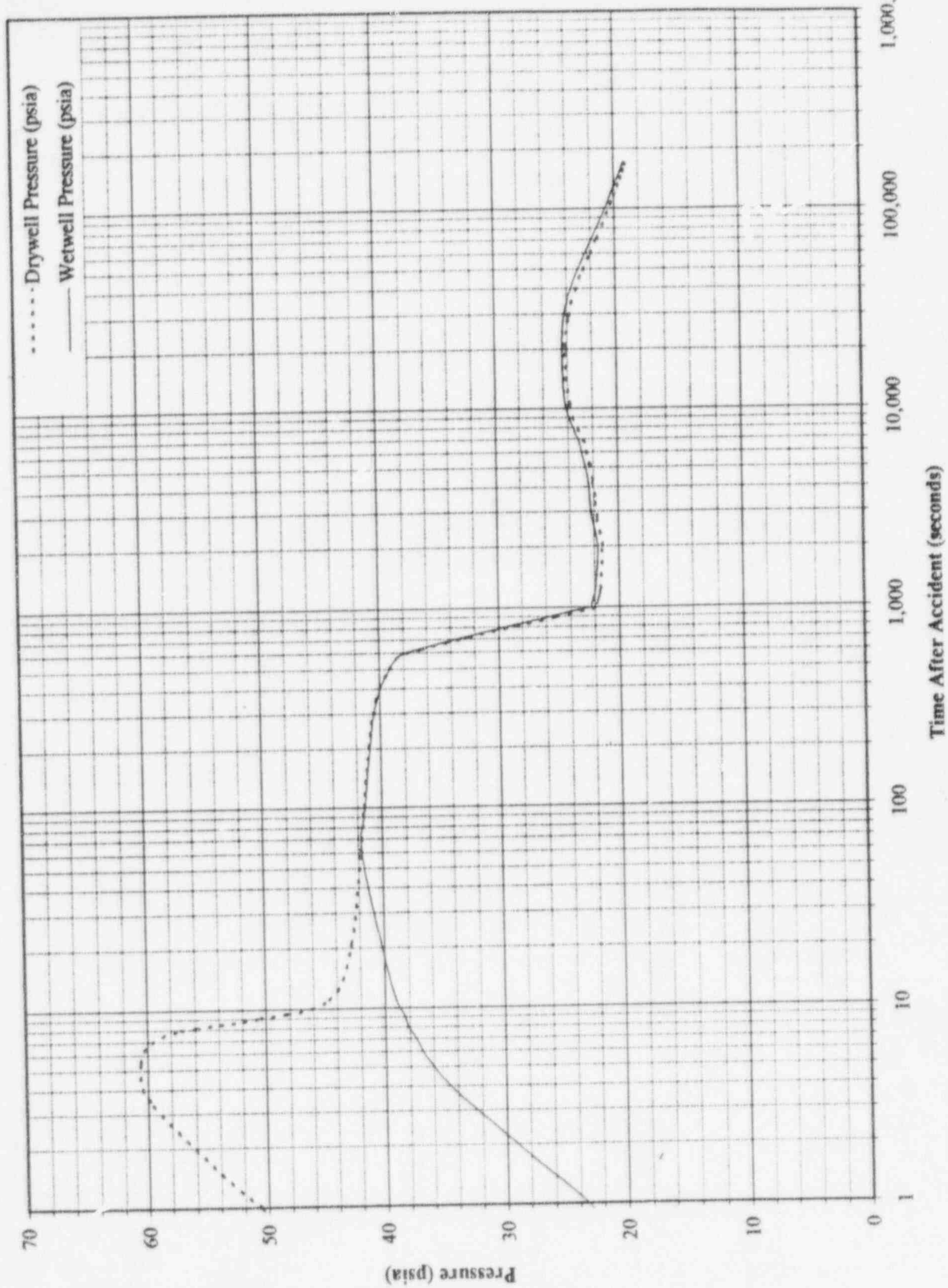


Figure 14.5-16 Design Basis Loss of Coolant Accident - Containment Pressure Response
75°F SSW Case

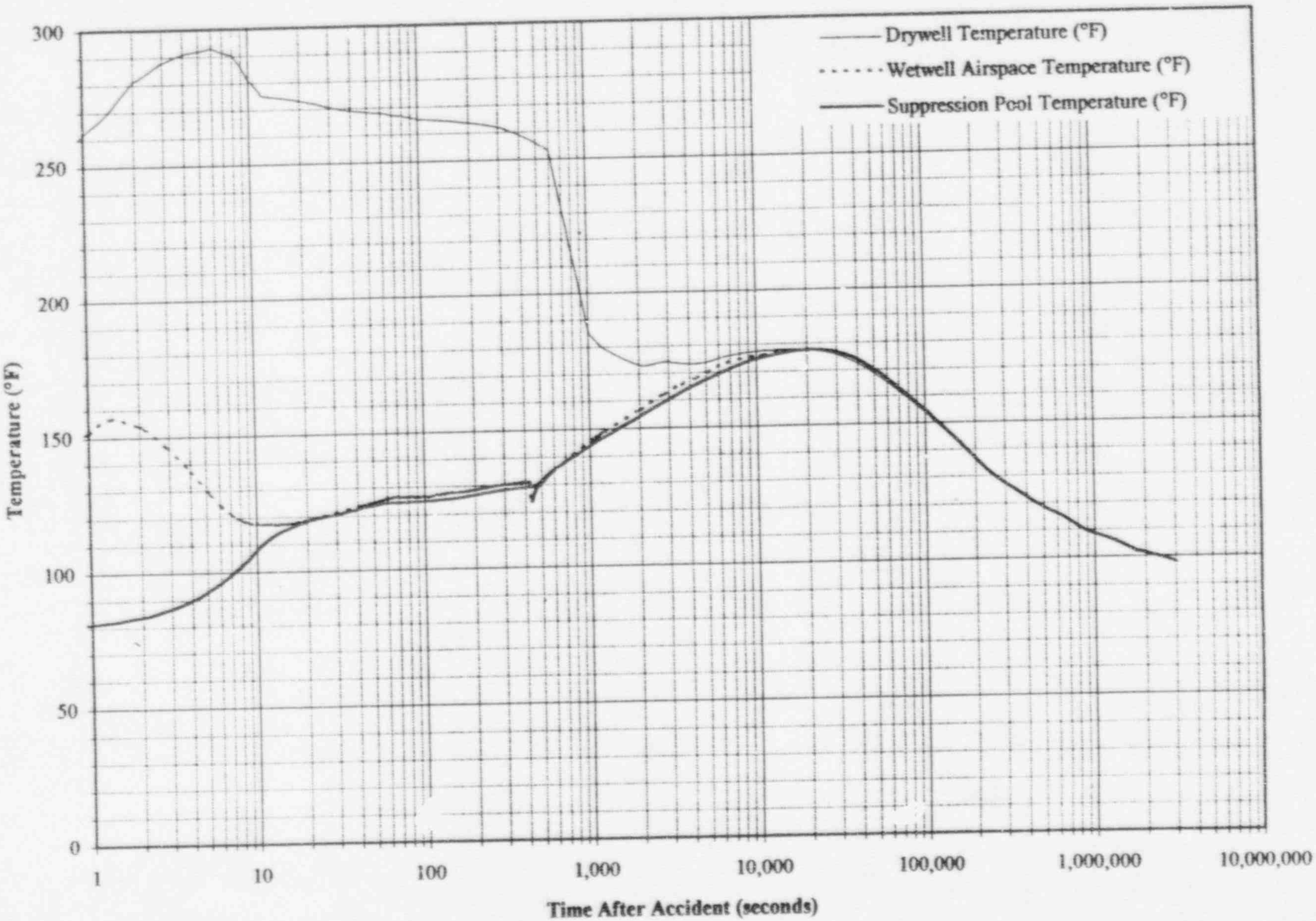
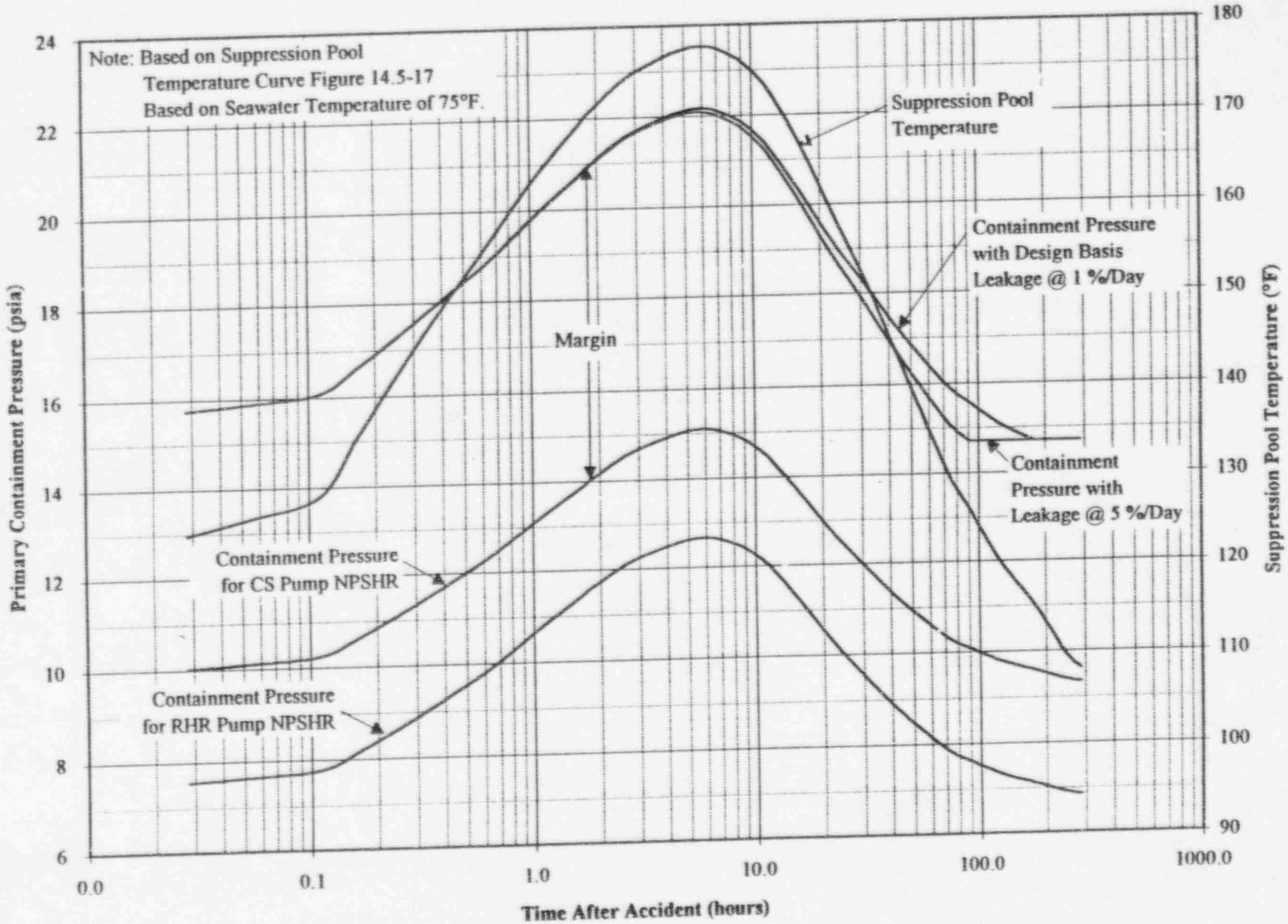


Figure 14.5-17 Design Basis Loss of Coolant Accident - Containment Temperature Response 75°F SSW Case



FSAR Figure 14.5-18 NPSH Availability for RHR and Core Spray System After a DBA-LOCA

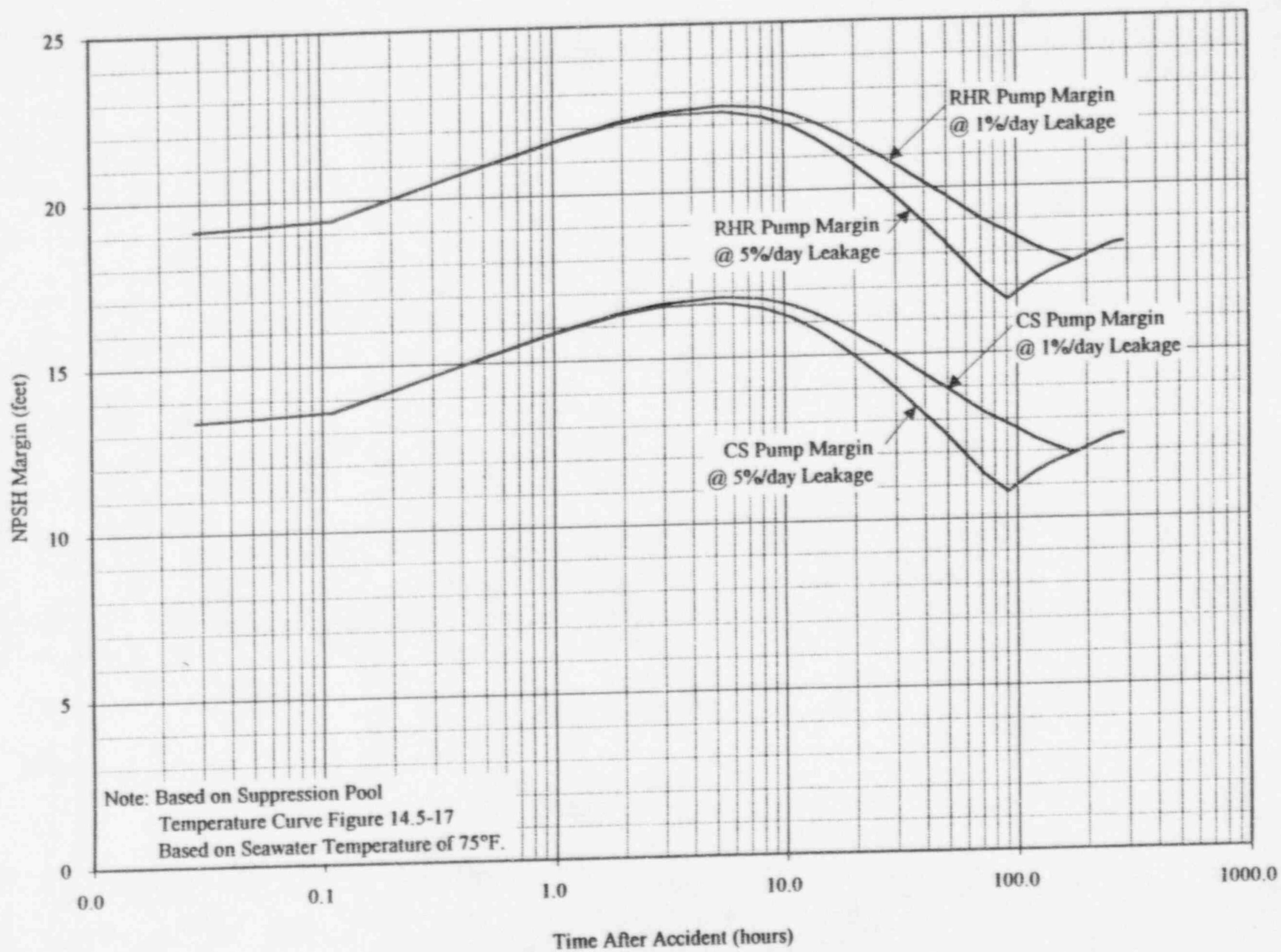


Figure 14.5-19 NPSH Margin for RHR and Core Spray Pumps After a DBA-LOCA

rate of the RHRS exceeds the energy addition rate from the decay heat, the containment pressure and temperature begin to decrease. Table 14.5-1 summarizes the peak containment pressure following the initial blowdown peak, the peak suppression pool temperature, and a summary of the equipment capability assumed in the analysis.

Case B

This case assumes that one RHRS loop is operating at design heat removal capacity (one RHR heat exchanger, one RHR pump, and design value of cooling water flow to one RHR loop operating in the suppression pool cooling mode). As in the previous case, there is no containment spray operation and the suppression pool cooling mode is assumed to be activated at 600 sec after the accident. The containment pressure response to this set of conditions is shown as curve "b" on Figure 14.5-5. The corresponding drywell and suppression pool temperature responses are shown as curves "b" on Figures 14.5-6 and 14.5-7. A summary of this case is shown on Table 14.5-1, including a summary of the equipment capability assumed in the analysis.

Case C

This case assumes the same equipment operability as Case B except that a portion of the discharge from the RHR heat exchanger is routed to the containment spray header. The remaining portion returns to the suppression pool via the suppression pool bypass line. It assumed that the containment spray is established at 600 sec after the accident.

The containment response to this set of conditions is shown as curve "c" on Figure 14.5-5. The corresponding drywell and suppression pool temperatures are shown as curves "c" on Figures 14.5-6 and 14.5-7. A summary of this case is shown on Table 14.5-1, including a summary of the equipment capability assumed in the analysis.

Comparing the "containment spray" Case C with the "no spray" Case B, it is seen that the suppression pool temperature response is the same because the same amount of energy is removed from the pool via the RHR heat exchanger. The total flow rate through the RHR heat exchanger is the same for Case B & C. However, the post blowdown containment pressure is higher for the "no spray" case, as shown by Figure 14.5-5. This, however, is of no consequence since the pressure is still much less than the containment design pressure of 56 psig. Figure 14.5-8 illustrates the slight effect on calculated containment leakage rate, due to the higher pressure.

Insert section for 75°F SSW Case, see item 17.

14.5.3.1.3 Core Standby Cooling System Pump Net Positive Suction Head

To assure proper operation of the RHRS circulating and reactor CSCS pumps following a design basis LOCA, precautions are taken to ensure that a net positive suction head (NPSH) margin is available to all above pumps at all times.

Replace section 14.5.3.1.3, see item 20.

The following conservative model is employed to make certain that this basic design requirement is met for all expected pressure pool temperatures.

1. The primary containment is assumed to contain the minimum credible mass of noncondensible gas prior to the design basis LOCA. (0 psig, 100 percent rh, and 150°F in drywell and 0 psig, 100 percent rh, and 80°F in wetwell.)

2. Containment leakage is calculated based on leak rate of 5 percent of the free volume per day which is 10 times the design basis leakage of 0.5 percent of the free volume per day.

3. Maximum initial torus pressure suppression chamber pool temperature is assumed (80°F).

4. Maximum Station Services Water System temperature is assumed (65°F).

5. Minimum Primary Containment System cooling capability, (e.g., diesel generation operation of one RHRS loop and two RHRS pumps) is assumed.

6. Minimum water volume in the torus suppression chamber pool is assumed (5.2×10^6 lb).

7. Drywell free volume temperature equal to pool torus pressure suppression chamber temperature following the accident. This assumption is discussed in detail as follows.

This model supplies a conservative lower bound on the available NPSH. However, additional significant margin is provided, as discussed.

At rated flow, the RHRS circulating and reactor CSCS pumps will require 28 ft or less of NPSH at 130°F. At higher torus pressure suppression chamber pool temperatures, the required NPSH will be less, but exactly how much less is not known at this time. Therefore, the discussion which follows will be predicated on the assumption that 28 ft of NPSH will be required by the RHRS and the Reactor Core Cooling System pump at rated flow for all pool temperatures.

Based on the above conservative model, the total NPSH available on Pilgrim Nuclear Power Station has been evaluated. The results are shown on Figure 14.5-9 as a function of pool temperature for both design basis Primary Containment System leak rate and 10 times the design basis leak rate. As can be seen, a large margin exists for all pool temperatures. The same information is presented in a different format on Figure 14.5-10. Here, the containment pressure is shown as a function of time. Also shown is the Primary Containment System pressure required to provide a total NPSH to the subject pump of 28

Replace with new text, see item 20.

Replace with new text see item 20.

ft. As can be seen, substantially more primary containment system pressure will be available than is required, based on the very conservative model. Therefore, it can be concluded that substantial NPSH margin will be available at all times following a design basis LOCA at Pilgrim Nuclear Power Station Unit 1.

During the Reactor Core Isolation Cooling System (RCICS) operation, the drywell free air volume cooler will normally remain operational. Due to the reduced heat load on the air coolers caused by the shutdown of the two Reactor Coolant Recirculation System pumps, the drywell temperature could actually be less than the normal operating value in spite of the fact that some of the air cooler capacity may also be shut down. The lower drywell temperature would tend to reduce the primary containment pressure which would reduce the NPSH available. In order to arrive at a conservative lower bound on the total NPSH available, the following model was assumed.:

1. No leakage from the primary containment (even at 5 percent free volume per day, leakage would be negligible during the short time period being considered).

2. Drywell and wetwell pressure equal (maintained equal by the vacuum breakers between the wetwell and drywell).

3. Torus air temperature equal to pool water temperature.

4. Drywell temperature during reactor core isolation cooling equal to 110°F, 20 percent rh (very conservative estimates).

5. Initial drywell conditions: 150°F, 0 psig, 100 percent rh

Actually, Assumption 4 and 5 are contradictory. If the heat load during normal operation is large enough to cause a drywell temperature of 150°F and a relative humidity of 100 percent, the air coolers would not be capable of reducing the drywell temperature to 110°F during RCICS operation. Such a heat load implies a small steam leak from the primary system.

The results are shown on Figure 14.5-11 as a function of pool temperature.

During RCICS operation, the pool temperature will never exceed 130°F, so that adequate NPSH will always be available. Even if the Reactor Primary System is depressurized (cooled down) following RCICS operation, the pool temperature will never exceed 163°F so that the margin in available NPSH will never be less than 4 1/2 ft. The 163°F temperature quoted here is the correct value at the end of reactor depressurization. Higher values of depressurized pool temperatures reported previously did not reflect the latest, larger heat exchangers and lower initial torus pressure suppression chamber pool temperature (80°F).

14.7 REFERENCES

1. "General Electric Standard Analysis for Reactor Fuel", NEDE-24011-P-A, Revision Number Listed in Latest Supplemental Reload Submittal in Appendix Q.

2. Moody, F.J., Maximum Flow Rate of a Single Component Two-Phase Mixture. Journal of Heat Transfer, Trans. ASME, Series C, Volume 87, p. 134.

3. Robbins, C.H., Test of a Full Scale 1/48 Segment of Humboldt Bay Pressure Suppression Containment. GEAP-3596, November 1960.

4. Moody, F.J., Two Phase Vessel Blowdown from Pipes. Journal of Heat Transfer, ASME Volume 88, pp. 285, August 1966.

5. Parsly, L.F., Design Consideration of Reactor Containment Spray System -- Part IV; Calculations of Iodine-Water Partition Coefficients. ORNL-TM-2412, Part IV, January 1970.

Add Reference 6. See item 21.

Technical Specification Bases Change

Change to Technical Specification Bases 3/4.7.A:

1. Replace the following sentences located in the first paragraph [see markup]:

   "Experimental data indicates that excessive steam condensing loads can be avoided if the peak local temperature of the pressure suppression pool is maintained below 200°F during any period of relief-valve operation with sonic conditions at the discharge exit. Analysis has been performed to verify that the local pool temperature will stay below 200°F and the bulk temperature will stay below 160°F for all SRV transients.

   with the following:

   "The maximum permissible bulk suppression pool temperature of 120°F is permissible since a complete accident blowdown can be accommodated without exceeding the bulk suppression pool temperature limit of 170°F immediately after blowdown. This 170°F LOCA blowdown limit is not a limit for the heatup of the suppression pool after the vessel is depressurized."

2. Add the following statement to the end of paragraph one:

   Current Technical Specification limits on suppression pool temperature ensure bulk pool temperature remains within an acceptable range to condense steam discharged to the suppression pool during a LOCA or SRV actuation.

A. Primary Containment (Cont)

Replace with item 1

Experimental data indicates that excessive steam condensing loads can be avoided if the peak local temperature of the pressure suppression pool is maintained below 200°F during any period of relief-valve operation with sonic conditions at the discharge exit. Analysis has been performed to verify that the local pool temperature will stay below 200°F and the bulk temperature will stay below 160°F for all SRV transients. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high pressure suppression chamber loadings. Add item 2

In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling heat exchangers, (3) initiate reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and frequently logged during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

If a loss-of-coolant accident were to occur when the reactor water temperature is below approximately 330°F, the containment pressure will not exceed the 62 psig code permissible pressure, even if no condensation were to occur. The maximum allowable pool temperature, whenever the reactor is above 212°F, shall be governed by this specification. Thus, specifying water volume-temperature requirements applicable for reactor-water temperature above 212°F provides additional margin above that available at 330°F.