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SEP REVIEW

OF

SAFE SHUTDOWN SYSTEMS

FOR THE

HADDAM NECK PLANT

309 260

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1.0 INTRODUCTION

The Systematic Evaluation Program (SEP) review of the "safe shutdown" subject encompassed all or parts of the following SEP topics, which are among those identified in the November 25, 1977 NRC Office of Nuclear Regulation document entitled "Report on the Systematic Evaluation of Operating Facilities":

1. Residual Heat Removal System Reliability (Topic V-10.B)
2. Requirements for Isolation of High and Low Pressure Systems (Topic V-11.A)
3. RHR Interlock Requirements (Topic V-11.B)
4. Systems Required for Safe Shutdown (Topic VII-3)
5. Station Service and Cooling Water Systems (Topic IX-3)
6. Auxiliary Feedwater System (Topic X)

The review was primarily performed during an onsite visit by a team of SEP personnel. This onsite effort, which was performed during the period July 11-13, 1978, afforded the team the opportunity to obtain current information and to examine the applicable equipment and procedures.

The review included specific system, equipment and procedural requirements for remaining in a hot shutdown condition (reactor shutdown in accordance with technical specifications, temperature between 200°F and 350°F) and for proceeding to a cold shutdown condition (temperature less than 200°F). The review for transition from operating to hot shutdown considered the requirement that the capability exists to perform this operation from outside the control room. The review was augmented as necessary to assure resolution of the applicable topics, except as note below:

Topic V-11.A (Requirements for Isolation of High and Low Pressure Systems) was examined only for application to the Residual Heat Removal (RHR) system. Other high pressure/low pressure interfaces were not investigated.

Topic IX-3 (Station Service and Cooling Water Systems) was only reviewed to consider redundancy and seismic and quality classification of cooling water systems that are vital to the performance of safe shutdown system components.

The criteria against which the safe shutdown systems and components were compared in this review are taken from the: Standard Review Plan (SRP) 5.4.7, "Residual Heat Removal (RHR) Systems;" Branch Technical Position RSB 5-1, "Design Requirements of the Residual Heat Removal System;" and Regulatory Guide 1.139, "Guidance for Residual Heat Removal." These documents represent current staff criteria for the review of applications for operating licenses.

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This comparison of the existing systems against the current licensing criteria led naturally to at least a partial comparison of design criteria, which will be input to SEP Topic III-1, "Classification of Structures, Components and Systems (Seismic and Quality)."

As noted above, the six topics were considered while neglecting possible interactions with other topics and other systems and components not directly related to safe shutdown. For example, Topics II-3.B (Flooding Potential and Protection Requirements), II-3C (Safety-Related Water Supply), III-4.C (Internally Generated Missiles), III-5.A (Effects of Pipe Break on Structures, Systems, and Components Inside Containment), III-6 (Seismic Design Considerations), III-10.A (Thermal-Overload Protection for Motors of Motor-Operated Valves), III-11 (Component Integrity), III-12 (Environmental Qualification of Safety-Related Equipment) and V-1 (Compliance with Codes and Standards) are among several topics which could be affected by the results of the safe shutdown review or could have a safety impact upon the systems which were reviewed. These effects will be determined by later review. This review did not cover, in any significant detail, the reactor protection system nor the electrical power distribution system both of which will be reviewed later in the SEP.

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The major factor in assessing the safety margin of any of the SEP facilities depends upon the ability to provide adequate protection for postulated Design Basis Events (DBEs). The SEP topics provide a major input to the DBE review, both from the standpoint of assessing the probability of certain events and that of determining the consequences of events. As examples, the safe shutdown topics pertain to the listed DBEs (the extent of applicability will be determined during the DBE review for Haddam Neck):

Topic	DBE Group*		Impact Upon Probability or Consequences of DBE
V-10.B	VII	(Spectrum of Loss of Coolant Accidents)	Consequences
V-11.A	VII	(Defined above)	Probability
V-11.B	VII	(Defined above)	Probability
VII-3	All	(Defined as a generic topic)*	Consequences
IX-3	III	(Steam Line [illegible]eak Inside Containment (Steam Line Break Outside Containment)	Consequences
	IV	(Loss of AC Power to Station Auxiliary) (Loss of all AC Power)	Consequences
	V	(Loss of Forced Coolant Flow) (Primary Pump Rotor Seizure) (Primary Pump Shaft Break)	Probability

*For a listing of DBE groups and generic topics, see Reference 12.

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Topic	DBE Group	Impact Upon Probability or Consequences of DBE
	VII (Defined above)	
X	II (Loss of External Load)	
	(Turbine Trip)	
	(Loss of Condenser Vacuum)	
	(Steam Pressure Regulator Failure) [closed])	
	(Loss of Feedwater Flow)	
	(Feedwater System Pipe Break)	
	III (Defined above)	Consequences
	IV (Defined above)	Consequences
	V (Defined above)	Consequences
	VII (Defined above)	Consequences

The completion of the safe shutdown topic review (limited in scope as noted above) provides significant input in assessing the existing safety margins for the Haddam Neck Plant.

2.0 DISCUSSION

2.1 Normal Plant Shutdown and Cooldown

A series of five operating procedures is employed to conduct a shutdown from full power to cold shutdown. The first procedure is Normal Operating Procedure (NOP) 2.2-1 Revision 6, "Changing Plant Load." NOP 2.2-1 is used to operate the plant above a minimum load of approximately 80 Mwe. The reactor controls are placed in manual and the load decrease is started by reducing steam flow with the turbine governor and manual insertion of the control group of rods. During the load decrease, plant parameters are maintained as noted below:

1. T_{avg} and pressurizer level are being maintained in accordance with the pressurizer level program (25-50%) by the chemical and volume control system (CVCS).

2. The pressure control system maintains the pressurizer and reactor coolant system at the normal operating pressure of 2000 ± 25 psig.

3. The feedwater control system maintains the steam generator water levels within the normal operating range of 25-50%.

4. Boration is performed as required to keep the control rods above the minimum position required by Technical Specifications.

When the power is reduced to 275-300 Mwe, a feedwater pump is taken out of service and, at 230-250 Mwe, a condensate pump is taken out of service. Seal water injection pumps are started to supply seal water to the main feedwater pumps seals when the Δp between condensate pump discharge and feedwater pump suction is less than 50 psi. Feedwater control is placed in the manual mode before minimum load is reached.

The next phase of the shutdown utilizes NOP 2.3-1 Rev. 0 "Minimum Load to Hot Standby." The plant conditions at the start of this phase are:

1. The plant is at minimum load at approximately 80 Mwe.

2. Steam generators are being maintained at normal operating level of 25-50% on narrow range with feedwater control in manual.

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3. Steam dump control is in automatic and set at 910 psig.

4. T_{avg} is being maintained at approximately 533°F by manual adjustment of the control group rod position within the desired operating range to compensate for reactivity changes.

The following steps are completed to put the unit in hot standby with the reactor critical:

1. Reset steam dump pressure controller (MS-PIC-1203) to 880-890 psig.

2. Reduce generator load and reactor power below P-7 permissive setting by gradual closing of turbine control valves and by insertion of control rods. Maintain T_{avg} of 533 ± 2°F during this operation.

3. Transfer power supplies for buses 1-1A and 1-1B from generator to outside lines. This places the reactor coolant pumps on off-station power.

4. Unload and separate the generator from the system in accordance with NOP 2.16-2 "Generator Phasing and Unloading."

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5. Open the feedwater regulating bypass valves and close the feedwater regulating valves.

6. Insert control rods until power is at 1×10^{-7} amperes on the Intermediate Range indicators.

The reactor is then shutdown by inserting the control rods to the 10 step limit and then opening the trip breakers per NOP 2.3-2 "Reactor Shutdown."

It is, also, possible to reach hot standby with automatic action of the reactor trip system or manual reactor trip action by the operator. These actions and the necessary manual actions of the operator are described in Emergency Operating Procedure (EOP) 3.1-1 Rev. 0 "Emergency Shutdown."

Operation at hot standby can be maintained in accordance with NOP 2.3-3 Rev. 1 "Operation at Hot Standby-Reactor Shutdown." The following conditions are maintained while operating at hot standby:

1. T_{avg} = 533 ± 2°F and RCS pressure is 2000 ± 25 psig.

2. Pressurizer levels is being maintained at normal operating range of 25 - 50%.

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3. Volume control tank level is within the normal operating range of 30 - 55%.

4. At least one reactor coolant pump is operating to maintain T_{avg} of 533 ± 2°F.

NOTE:

The Reactor Coolant Pump in operation must be No. 3 or 4 to provide sufficient head to operate the pressurizer sprays.

Steam generator liquid levels are manually maintained at normal operating range of 25 - 50% on narrow range indicator.

5. One Auxiliary Feedwater Pump or a main steam generator feed pump is in operation and feeding the steam generators via the feedwater regulating bypass valves, FW-HICV-1301-1, 2, 3 & 4.

6. Steam Generator may be dumping excess steam to condensers via steam dump or venting to atmosphere via atmospheric vent. (MS-HICV-1201).

7. Reactor Coolant System is borated to the required boron concentration to maintain a 3% shutdown margin with all rods inserted.

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8. Letdown purification is operating in accordance with requirements of the Plant Chemistry Department.

9. Steam generator non-return bypass valves MS-NRV-17, 27, 37 and 47 are closed.

10. Sufficient water is available in the demineralized water storage tank (DWST).

The Technical Specifictions require a minimum of 50,000 gallons of water in the DWST and a minimum of 80,000 gallons of water in the primary water storage tank (PWST). Water may be transferred from the PWST to the DWST at the rate of 200 gpm with tranfer pumps. A third source of water is available from the Recycle Primary Water Storage Tank (RPWST). There are no Technical Specification requirements on the amount of water in the RPWST; however, the licensee normally, maintains 95,000 to 100,000 gallons stored in this tank.

If the method of heat removal is through the atmospheric vents, the source of water for the auxiliary feedwater pumps is the DWST. If the steam dumps are used, the level of water in the hotwell will be used and the excess water is directed through the condensate pumps to the DWST and then back to the auxiliary feedwater pumps.

The procedure to cool the unit to cold shutdown is NOP 2.3-4 Rev. 5 "Hot Standby to Cold Shutdown."

The reactor coolant system must be borated to the cold shutdown or refueling boron concentration before cooldown is initiated. Makeup for coolant contraction must be at the same concentration as that required for cold or refueling shutdown.

The pressurizer heaters are turned off and all but one reactor coolant pump are shutdown. If it is necessary to shutdown both pumps No. 3 and 4, spray to the pressurizer is delivered through the auxiliary spray line from the charging line. The letdown and charging flows are adjusted to account for the reduced pressure drop across the letdown orifices and for the thermal contraction of the reactor coolant.

When the pressurizer temperature reaches approximately 450°F and the pressure is approximately 600 psig, the steam bubble is collapsed in the pressurizer and the pressurizer is filled.

The next step in the plant cooldown is to energize the low temperature overpressurization relief isolation valves RP-MOV-596, PR-MOV-597, PR-MOV-598, PR-MOV-599. When RCS temperature reaches 340°F, pressurizer temperature <430°F, reduce RCS pressure to 350 psig. Place the low temperature overpressurization relief valves PR-RV-587 and PR-RV-588 in service by opening motor operated isolation valves PR-MOV-596, PR-MOV-597, PR-MOV-598 and PR-MOV-599.

NOTE: PR-RV-587 and PR-RV-588 are set to relieve at 380-psig.

When the reactor pressure has been reduced to 300 psig the residual heat removal (RHR) system is placed in service; and when steam pressure is at 300 psig, the auxiliary feedwater pump is taken out of service and feedwater can be supplied to the steam generators with the condensate pumps.

The cooldown is continued with the RHR system until the reactor coolant system meets the cold shutdown requirements. The cooldown procedure calls for intermittent operation of a single reactor coolant pump which requires repetitive operation of the RHR isolation valves during the cooldown.

The RHR system has two one-half size heat exchangers and two one-half size pumps (FSAR 5.2.3.3). There are two methods of removing heat from the RHR heat exchangers. The preferred method for decay heat removal is to use the component cooling system and then transfer the heat from the component cooling water to the service water system; the second method is to introduce service water directly to the secondary of the RHR heat exchangers. The use of the second method is called for in Emergency Operating Procedures (EOP) 3.1-11 "Loss of Component Cooling."

The normal operating pressures of the systems used for shutdown cooling are 150 psig plus the pump suction for the RHR loop, 82 psig for the component cooling loop, and 55 to 70 psig for the service water system; therefore, the flow of impurities would be away from the reactor coolant system.

Overpressurization of the RHR system from the RCS is avoided by administrative control of two locked isolation valves and by a pressure interlock on two more isolation valves. A relief valve is installed on the RHR system that relieves at 500 psig and has a 960 gpm relieving capacity.

2.2 Shutdown and Cooldown with a Loss of Offsite Power

The shutdown following a loss of offsite power is achieved with EOP 3.1-9 Rev. 4 "Total Loss of AC." A station blackout results in the loss of circulating water pumps, the main feedwater pumps and condensate pumps; the reactor coolant pumps remain in service for approximately one minute and then undergo a pump coastdown that is extended by the intertia of the fly wheels on the pumps. The pump coastdown time is estimated to be three to four minutes with no pumps running, i.e., no back flow, this estimate is given in the March, 1968 report on the natural circulation test "Natural Circulation Test of Reactor Coolant System."

On loss of offsite power, the plant emergency diesel generators start automatically.

The operator is directed to restore diesel generator power to certain vital equipment and to start the auxiliary feed pump to feed the steam generators through the feedwater bypass valves. The atmospheric dump valve and hogging air ejectors are used to control T_{avg} in the reactor coolant system. Emergency Operating Procedures EOP 3.1-9 does not contain instructions to attain and maintain cold shutdown conditions during a loss of offsite power. EOP 3.1-9 does not contain instructions on the source of water after the supply in the DWST is exhausted. The site experienced loss of offsite power on April 27, 1968, July 15, 1969 and June 26, 1976; the unit

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was on line when the first two events occurred and was in the refueling mode during the last event.

3.0 CONFORMANCE WITH BRANCH TECHNICAL POSITION 5-1 FUNCTIONAL REQUIREMENTS

The current NRC criteria used in the evaluation of the design of the systems required to achieve cold shutdown for a new facility are listed in Standard Review Plan (SRP) 5.4.7 and Branch Technical Position RSB 5-1. The following paragraphs give a point by point comparison of Branch Technical Position RSB 5-1 to the shutdown systems at the Haddam Neck Plant.

Branch Technical Position

"A. Functional Requirements

"The system(s) which can be used to take the reactor from normal operating conditions to cold shutdown shall satisfy the functional requirements listed below.

1. The design shall be such that the reactor can be taken from normal operating conditions to cold shutdown using only safety-grade systems. These systems shall satisfy General Design Criteria 1 through 5.

2. The system(s) shall have suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system function can be accomplished assuming a single failure.

3. The system(s) shall be capable of being operated from the control room with either only onsite or only offsite power available with an assumed single failure. In demonstrating that the system can perform its function assuming a single failure,

limited operator action outside of the control room would be considered acceptable if suitably justified.

4. The system(s) shall be capable of bringing the reactor to a cold shutdown condition, with only offsite or onsite power available, within a reasonable period of time following shutdown, assuming the most limiting single failure."

The capability of the safe shutdown systems for Haddam Neck to meet these criteria is discussed below.

3.1 Background

These requirements are stated with respect to plant shutdown and cooldown with only offsite or only onsite power available. The staff evaluated the plant's ability to conduct a shutdown with only offsite power available and determined that the only onsite power available case is more limiting. The plant electrical system is sufficiently versatile to allow energizing of all necessary equipment from only offsite power. Therefore, the staff concentrated its evaluation of the safe shutdown systems to the shutdown following a loss of offsite power.

A "safety grade" system is defined, in the NUREG 0138 (Reference 1) discussion of issue #1, as one which is designed to seismic Category I (Regulatory Guide 1.29), quality group C or better (Regulatory Guide 1.26), and is operated by electrical instruments and controls that meet Institute of Electrical and Electronics Engineers Criteria

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for Nuclear Power Plant Protection Systems (IEEE 279). The Haddam Neck plant received its Provisional Operating License on June 30, 1967 and its Full Term Operating License on December 27, 1974, so that plant was designed and constructed prior to the issuance of Regulatory Guides 1.26 and 1.29 (as Safety Guides 26 and 29 on March 23, 1972 and June 7, 1972 respectively). Also, proposed IEEE 279, dated August 30, 1968, was not used in the design of instrumentation and control systems at Haddam Neck. In addition, the Haddam Neck plant was built and licensed prior to the issuance of the proposed General Design Criteria on July 11, 1967. Therefore, for this evaluation, systems which should be "safety grade" are the systems identified in Table 3.1 and in the following list of minimum safe shutdown systems.

General Design Criterion (GDC) 1 requires that systems important to safety be designed, fabricated, erected, and tested to quality standards, that a Quality Assurance (QA) program be implemented to assure these systems perform their safety functions, and that appropriate records of design, fabrication, erection, and testing are kept.

Regulatory Guide (RG) 1.26 provides the current NRC criteria for quality group classification of safety-related systems. Table 3.1 provides a comparison of the Haddam Neck safety grade shutdown

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systems with RG 1.26. Although RG 1.26 was not in effect when Haddam Neck was constructed, the licensee has since classified the systems in accordance with this guide (Reference 1). Even though the safety-related systems were not designed, fabricated, erected, and tested using RG 1.26, the maintenance and repair of the classified systems is currently conducted in accordance with this guide.

In Reference 2, the licensee has identified maximum seismic ground accelerations which were used in the design of structures, systems, and components important from the standp of nuclear safety. The reactor coolant system and high and l ressure safety injection systems have been designed for a 0.17 g maximum ground acceleration, and the remaining structures and elements of the plant are cons ered capable of withstanding the seismic forces corresponding to a ground acceleration of at least 0.03 g. No structures or equipment are classified as seismic category I per Regulatory Guide 1.29. Therefore, in Table 3.1, a ground acceleration level will be stated instead of a seismic design classification.

At the time the Haddam Neck plant was originally licensed, the NRC (then AEC) criteria for QA were not developed. However, the QA program for plant operations was reviewed by the staff and found to be in conformance with 10 CFR 50, Appendix B (Reference 3). Appropriate records concerning design, fabrication, erection and testing of equipment important to safety are maintained by the licensee in accordance with the QA program and the plant Technical Specifications.

GDC 2 states that structures and equipment important to safety shall be designed to withstand the effects of natural phenomena without loss of capability to perform their safety function. Natural phenomena considered are: hurricanes, tornadoes, floods, tsunami, seiches and earthquakes.

Measures were taken in the design of the plant to protect against floods and earthquakes. During the Full Term Operating License review, the staff agreed with the licensee's conclusions in Reference 4 that the effects of tornadoes, floods, earthquakes, winds, ice, and other local site effects on structures and equipment important to safety were acceptable (Reference 5).

The effects of tornadoes will be reevaluated during the course of the SEP in Topics II-2.A "Severe Weather Phenomena," III-2 "Wind and Tornado-Loadings," and III-4.A "Tornado Missiles." The effects of flood will be reassessed in the SEP review under Topics II-3.B "Flooding Potential and Protection Requirements" and III-3 "Hydrodynamic Loads." And within the SEP review, the potential for and consequences of a seismic event will be reassessed under several review topics.

GDC 3 requires structures, systems, and components important to safety to be designed and located to minimize the effects of fires and explosions.

The Haddam Neck fire protection reevaluation resulting from the Browns Ferry fire is currently underway in the NRC Division of Operating Reactors. The fire protection Safety Evaluation Report was issued October 3, 1978. The results of this reevaluation will be integrated into the SEP assessment of the plant.

GDC 4 requires that equipment important to safety be designed to withstand the effects of environmental conditions for normal operation, maintenance, testing, and postulated accidents. Also the equipment should be protected against dynamic effects including internal and external missiles pipe whip, and fluid impingement.

The SEP will consider the various aspects of this criterion when reviewing topics III-12 "Environmental Qualification of Safety-Related Equipment," III-5.A "Effects of Pipe Breaks Inside Containment," III-5.B "Pipe Breaks Outside Containment," and III-4 "Missile Generation and Protection."

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GDC 5 is not applicable for the Haddam Neck plant because it does not share any equipment with other facilities.

In order to accomplish a plant shutdown and cooldown following a loss of offsite power, certain "tasks" must be performed such as core decay heat removal, steam generator makeup, and component cooling. The staff and licensee developed a "minimum list" of systems necessary to perform these tasks considering a loss of AC power and the most limiting single failure. The system were then evaluated with respect to their ability to perform those tasks, and the functional requirement of BTP 5-1.

The minimum systems (or components)* are given below:

1. Steam System ASME Code Safety Valves
2. Atmospheric Dump Valve (ADV) Steam Generator Vents, and other vent paths
3. Auxiliary Feed Pumps

*CYAPCO is evaluating the need for the Instrument Air System. The IAS may not be required since the air operated components necessary for safe shutdown may fail to their safe or required position upon a loss of air. Also, CYAPCO has stated that pressurizer heaters are required to maintain RCS pressure control during hot shutdown, and the subsequent RCS cooldown.

4. Water Sources - Demineralized Water Storage Tank (DWST), Primary Water Storage Tank (PWST) and Primary Water Transfer Pump
5. Residual Heat Removal System
6. Service Water System
7. Chemical and Volume Control System
8. High Pressure Safety Injection
9. Containment Fan Coolers (Cooled by Service Water System)
10. Pressurizer Power Operated Reliefs
11. Emergency Power Systems (AC and DC) for the above equipment
12. Instrumentation for the above equipment

The staff's evaluation of each of these systems with respect to the BTP 5-1 functional requirements is given below.

3.2 Functional Requirements

Steam System ASME Code Safety Valves

Task: Removal of core decay heat by automatically venting steam from the main steam system.

Discussion:

Decay heat is initially removed from the RCS by the automatic actuation of the main steam safety valves (MSSV). These valves are ASME code, self actuated valves that relieve independently and directly to atmosphere. Each steam generator has four MSSVs mounted on a short length of 24" OD piping connected to each main steam header. The size, setpoint and capacity of each MSSV is given below:

8" x 14"	P_{set} = 985 psig	$\approx$	594,000 lbm/hr
8" x 14"	P_{set} = 1015 psig	$\approx$	594,000 lbm/hr
8" x 14"	P_{set} = 1025 psig	$\approx$	594,000 lbm/hr
8" x 14"	P_{set} = 1034 psig	$\approx$	594,000 lbm/hr

Therefore, the total relieving capacity available to each steam generator is about 2.376×10^6 lbm/hr. Immediately after the loss of AC, turbine trip and reactor scram, the steam generator pressure and the RCS temperature (by natural circulation) will be controlled by the operation of 10 air operated turbine bypass valves (TBV) and the MSSVs. The TBVs will continue to relieve system pressure to the condensers as long as the condenser vacuum remains acceptable (the main circulating pump will be without power) and the IAS maintains air pressure to the valves. If the TBVs are not operable, the MSSVs will lift to control steam generator pressure.

Redundancy

As noted above, the TBVs will be available for steam relieving for some period following the loss of AC. CYAPCo has stated that during a previous loss of AC, the condenser vacuum was not lost for approximately 20 min due to gravity flow of the condenser cooling water. However, the staff gave no credit for the TBVs steam relieving flowrate into the condenser since the condenser vacuum following the loss of AC would depend on unquantifiable factors (river height, decay heat, plant power at time of trip, hotwell level at the time of the trip).

The staff calculated the number of MSSVs required to maintain the RCS temperature at an acceptable level following a scram from full power after a loss of AC. Based on an initial after scram decay heat level of 6.75%, the relieving rates shown above, and an energy removal capability of about 650 Btu/lbm (h_{fg} at P = 1000 psig), two 985 psig (P_{set}) MSSVs will maintain RCS temperature (at about 540°F) immediately after the loss of AC and scram.* Even though the MSSVs are passive devices, and as such are normally considered failure free, the staff considers 3 MSSVs necessary to maintain RCS temperature following the event.

Location and Operation

The staff evaluated the equipment discussed above with respect to its location and operability during a loss of offsite AC. The MSSVs are located in the main steam and feedwater penetration enclosure.

*This quantity (2 MSSVs) does not consider any pressure transients on the steam system as a result of the loss of load.

They are self actuated and require no electrical power for operation. They cannot be manually operated.

Atmospheric Dump Valve Steam Generator 1" Vents, and Other Vent Paths

Task: Removal of core decay heat by venting steam from the main steam system to atmosphere.

Discussion:

Immediately after the loss of offsite AC, turbine trip and reactor scram, the MSSVs automatically actuate to control steam system pressure and RCS temperature. However, the Haddam Neck Turbine Bypass Controller (TBC) provides additional steam relieving paths to (1) prevent and/or limit the operation of the MSSVs* and (2) provide one means of RCS cooldown by venting steam to the main condenser. Several other steam relieving paths are available for RCS cooldown.

The air controlled Atmospheric Dump Valve (ADV) vents steam from any or all of the four 24" (OD) steam lines via a decay heat release header (DHRH). The DHRH is pressurized from the 24" steam lines

*Since the TBC is an active component, the staff considered the MSSVs as the principal components in initially controlling RCS temperature following the loss of AC. However, the TBC would normally act to prevent or limit MSSV operation.

via 3" (OD) lines just upstream of the non-return valves. The DHRH is normally pressurized, and supplies the two turbine driven auxiliary feedwater pumps as well as the ADV. The DHRH is located in the upper level of the steam and feedwater penetration enclosure immediately outside the reactor containment.

The ADV is a 3" (OD) valve which is operated by air pressure acting on a diaphram. The position of the ADV is controlled by the air pressure on the actuator and is controlled from the control room.

Since CYAPCO has not included any air systems in the minimum list of systems, we assume air pressure (control and service air) is lost following the loss of AC. Under this condition, the ADV would fail shut, and could not be manually operated.

Condenser vacuum is normally initially established by two "hogging jets" and maintained by two sets of steam jet air ejectors (SJAEs). The hogging jets are single stage venturi type air ejectors which use main steam supplied through a pressure reducing valve and normally draw noncondensible gases directly from the condenser steam space, and discharge to atmosphere.

Each SJAE has two first stage and two second stage nozzles. The first stage nozzles, which use reduced pressure main steam, normally draw noncondensible gases directly from the condenser steam space and discharge to the SJAE inter-condenser which is cooled by main condensate. The second stage nozzles also use reduced pressure main steam and draw from the inter-condenser and discharge to the after-condenser which is also cooled by main condensate. The condensed steam from the inter- and after-condensers is directed back to the main condenser, and the accumulated non-condensible gases are vented to atmosphere.

During the cooldown following a loss of AC, the SJAEs and the hoggers are used to remove energy from the steam generators by bleeding steam from the main steam lines. The normal suction paths to the main condenser would be isolated. The hoggers and the SJAEs are manually controlled devices and require adjustments as steam pressure varies.

The Auxiliary Feedwater Pumps (AFPs) are utilized to provide feedwater to the steam generators during the loss of AC, and are described in the following section. The AFPs are turbine driven pumps each rated at 430 hp at full flow. Thus, the AFPs are another means of removing steam generator (and RCS) energy by venting main steam through the turbine exhausting to the atmosphere. Each AFP turbine has a relief valve which opens at ___ psig, and relieves about 38,000 lbm/hr.

Upstream of each main steam non-return lines a 1" vent with two manually operated isolation valves is provided to permit steam system venting during RCS and steam system startups. These vents relieve directly to atmosphere in the steam and feedwater penetration enclosure. Thus, these four 1" vents provide another path for steam energy removal.

The licensee has noted that during a previous loss of AC, the condenser vacuum did not decay below 20" for 20 minutes. Apparently, the arrangement of the weirs relative to the river and the condenser heights allowed enough reverse flow through the condenser tubes to remove the energy from the steam flow via the turbine bypass valves. Although the staff and licensee could not quantify the reverse flow and resulting condenser cooling, this is certainly a potential technique for RCS energy removal following a loss of AC and scram.

The earliest time following the loss of AC and scram when each component, individually can remove the amount of core decay heat being added to the RCS is shown below. The staff used the core decay heat curve from ANS 5.1, the steam flow rates presented below, and an energy removal rate of 650 Btu/lbm (h_{fg} at P = 1000 psig).

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Component	Steam Flow (p 1000 psig)	RCS Energy Removal	Full Power Fraction	Corresponding Time after Scram
ADV	133,000 lbm/hr	86.4×10^6 Btu/hr	1.39%	$\approx$ 50 min
4 1" vents	162,700 lbm/hr	105.4×10^6 Btu/hr	1.69%	$\approx$ 25 min
2 AFPs	32,000 lbm/hr	20.7×10^6 Btu/hr	0.33%	$\approx$ 117 hours
Hoggers	9,300 lbm/hr	6.0×10^6 Btu/hr	0.10%	-
SJAEs	1,000 lbm/hr	0.7×10^6 Btu/hr	0.01%	-
AFP RVs	38,000 lbm/hr	24.6×10^6 Btu/hr	0.39%	$\approx$ 55 hours

The time when the component energy removal capability equals the decay heat input corresponds to the time when (1) plant cooldown commences if the component is used and (2) intermittent MSSV lifting would stop.

Redundancy

To establish the degree of redundancy provided by the equipment discussed above, the staff and licensee performed scoping calculations to determine the RCS cooldown times (e.g., time to go from 544°F to 350°F) using various combinations of the above components. The staff's calculations are presented below.

Component	RCS Cooldown Time[1,2]
ADV	425°F at 24.7 hrs
4 1" S/G vents	400°F at 25.4 hrs
ADV + 4 vents	15.8 hrs
4 vents + 2 AFPs + SJAEs + Hoggers	8.3 hrs

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Note 1: If the staff's calculations showed that the Tech Spec minimum auxiliary feedwater inventory was expended before the RCS was cooled to 350°F, then the temperature and time shown reflect the values when the 130,000 gal is consumed.

Note 2: The staff's calculations assumed no credit for the colder auxiliary feedwater's h_f, rather, the staff conservatively assumed the saturation h_f corresponding to the steam pressure. This is an extra conservatism of at least 350 Btu/lbm which would shorten the calculated cooldown times.

The staff also performed some scoping calculations to determine the ability of the RCS to be cooled after a 4 hr wait time at the hot shutdown condition. We found that the time to reach 350°F is about the same as if the cooldown had begun as soon as possible after the scram (with the 1" vents + 2 AFPs + SJAEs + Hoggers). From the table above, the RCS temperature is 350°F about 8.3 hours after the scram if the cooldown is initiated ASAP after the scram*. We calculated that if the cooldown were initiated at SCRAM + 4 hrs, the RCS temperature

*For the RCS cooldown by the steam relief from the 4 steam generator vents, 2 AFP turbines, 2 SJAEs, and the hoggers, the cooldown starts about 10 min after the scram.

is 350°F about 5.4 hours later, or 9.4 hours after the SCRAM. However, this resulted in a significant cooldown rate initially, although still less than the Technical Specification maximum (76.2°F/hr vice 100°F/hr). The staff also confirmed that one AFP has sufficient flow to keep the steam generator water level during all phases of this rapid cooldown.

CYAPCO has stated that the RCS has been cooled down from about 540°F to 350°F in 12-15 hours using (not simultaneously) the ADV, steam generator vents, 2 AFP (turbines), SJAEs and the Hogging jets. During the cooldown, the RCS flow was provided by 1 or more RCPs, which is an additional source of heat to the RCS.

Based on the staff's calculations, the licensee's experience, the availability of the two AFP relief valves,* and the water supplies to be discussed in a following section, we conclude that there are sufficient means to cool the RCS to the RHR cut-in point prior to the expenditure of the Tech Spec minimum 130,000 gal.

Location and Operation

The staff evaluated the equipment discussed above with respect to its location and operability during a loss of offsite AC. The

*The ADV may not be available following the loss of AC if the air system is considered unavailable.

table below gives the equipment's location, the places from where it may be operated, and the equipment's power supply.

Auxiliary Feedwater Pumps

Task: Providing steam generator makeup inventory whenever RCS temperature is ≥ 350°F.

Discussion:

While the RCS temperature is above 350°F, the core decay heat is removed by bleeding steam from the steam generators using the various components and flowpaths discussed previously. The condensate and feedwater pumps are normally powered from offsite power so these components will not be available. (The Emergency Diesel Generators have insufficient capacity to power these components following the loss of the station generator and offsite power.)

Each steam generator contains about 45,980 lbm of feedwater at full power, therefore, about 184,000 lbm are immediately available for primary system energy removal (by MSSV actuation).

The staff calculated the amount of energy 184,000 lbm would remove by vaporization. These calculations show that this inventory is sufficient to maintain RCS temperature acceptable without initiating any steam generator makeup, for approximately 1 hour. Even if

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EQUIPMENT	LOCATION	OPERATION	POWER SUPPLY
Atmospheric Dump Valve	Upper level of the steam and feedwater penetration enclosure	Control room and ____________ ____________	
Hogging Jets	Intermediate level of the turbine building, near generator and of turbine	Local manual operation only	No ejection power is needed
SJAEs	Intermediate level of the turbine bldg.	____________ ____________	No electrical power is needed
1" steam vents	Upper level of the steam and feedwater penetration enclosure	local manual operation only	No electrical power is needed
AFP turbines	(see discussion of AFP)	(see discussion of AFP)	(See discussion of AFP)
AFP turbine RVs	Adjacent to each AFP turbine	RVs are self-actuated when turbine casing pressure reaches ___ psig. The CR operator _____ cause the RVs to open, and the RVs can also be manually opened	No electrical power is needed

there were a 15 sec delay time between the loss of AC (loss of load and feedwater) and the reactor scram, (i.e., the steam generators are producing 100% power without feedwater), the staff calculation shows that steam generators can remove the core decay heat for about 55 minutes before boiling dry.

Two turbine driven Auxiliary Feed Pumps (AFP) provide the steam generators with feedwater following a loss of offsite AC. The AFPs are centrifugal pumps which take water from the demineralized water storage tank (DWST) and inject it into the four steam generators. A buried pipe connects the DWST to the suctions of both AFPs. The pumps discharge to a common header which branches into two parallel paths to the main steam generator feed system. One path enters the turbine building via a buried pipe and connects to the bypass lines around the four main feed regulating valves. The other line passes through the containment wall and connects to each steam generator main feed line downstream of the main feed non-return valve.

The AFP discharge header to the turbine building is equipped with a normally open manual valve in the main steam and feedwater penetration enclosure and a check valve in the turbine building. This line branches and connects to the four main feed regulating valve bypass lines upstream of the air operated main feed regulating valve bypass valves. The bypass valves fail open on a loss of air pressure.

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The AFP discharge header which enters containment from the main steam and feedwater penetration enclosure is equipped with a motor operated valve (MOV-35), which is normally controlled from the control room, and a check valve outside containment. Inside containment, the header branches into four supply lines, one for each steam generator. Each of these lines has a normally open manual isolation valve and a check valve. MOV-35 is powered from MCC #7 (bus 7-6) which can be energized from emergency diesel generator 2B. Also, MOV-35 can be manually operated with an installed handwheel should a failure disable electrical power to the valve.

The steam supply for the AFPs comes from the DHRH and the piping design is such that either AFP can receive steam from any one or all four steam generators. From the DHRH, the steam supply for each AFP passes through a normally open manual valve and an air operated control valve which is normally operated from the control room. The air operated valves fail shut on loss of air pressure, but a handwheel is provided on each valve to effect a local manual startup of an AFP.

Based on the previous discussion of after scram steam generator boil-off rate, sufficient time is available for an operator to manually start an AFP if the control air system were inoperable.

Redundancy

Each AFP delivers about 450 gpm of auxiliary feedwater to the steam generators. The staff calculated that this flow is sufficient to control and raise steam generator level about 30 sec after the scram.*

The auxiliary feed system (AFS) is capable o withstanding a single active component failure without the loss of its ability to perform its design function. However, the approximately 20 feet of discharge header shared by both AFPs and the single suction line from the CST render the system susceptible to a single passive failure which could disable both AFPs. Also the passive failure of the condensate supply line from the CST to several non-essential systems could divert condensate away from the AFPs since the AFP suction line connects directly to the non-essential service header from the CST.

If the AFS were disabled by one of the above passive failures, the licensee has described a method for removing core decay heat by blowing down the RCS to the primary containment, via the pressurizer power operated relief valves, and injecting coolant with the safety

*The calculation shows that the vaporization rate at 30 seconds is equal to the AFP mass input rate.

injection systems.* Long term cooling could be accomplished by either ECCS recirculation or by use of the RHR system (after RCS temperatures and pressures were lowered sufficiently for RHR initiation). Containment cooling would be provided by the Containment Recirculation Fan Coolers. An alternate method of feeding the steam generators, which is not permitted by procedures, involves blowing a steam generator completely dry and feeding by means of a low head pump (fire pump, condensate pump, etc.). The feasibility of these alternate decay heat removal methods will be pursued with the licensee.

Location and Operation

The staff evaluated the equipment discussed above with respect to its location and operability during a loss of offsite AC. The table below gives the equipment's location, the places from where it may be operated, and the equipment's power supply.

Water Source - DWST and PWST

Task: Provide water to the Auxiliary Feedwater System for steam generator makeup.

*This method is discussed in Reference 13. The systems involved have been included in the minimum list of safe shutdown systems in Section 3.1.

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EQUIPMENT	LOCATION	OPERATION	POWER SUPPLY
Auxiliary Feedwater Pumps (2)	Lower level of the steam and feedwater penetration area (north and south end)	Both pumps are operable from the CR using an air controller which positions the steam inlet valves. If air pressure is lost, the valves can be positioned manually.	No electrical power is needed
MOV 35 (single valve which isolates AFW from entering Containment - 1 path of AF flow)	Lower level of the steam and feedwater penetration area, above south pump	Control room and local manual	MCC #7(bus 7-6)
Auxiliary FRVs (4)	Feedwater Regulating valve [illegible] of turbine building	Control room (using air signal)	Control air, fail open on loss of air
Feedwater MOVs (must be shut)	Feedwater Regulating valve area of turbine building	Control room ______ ______ and ______ ______	
or Feedwater reg. valves must be shut	Feedwater Regulating valve area of turbine building	Control room	Control air, fail shut on loss of air

Discussion:

Both AFPs take a suction from the DWST via the 10" hotwell makeup and rejection line. This line leaves the bottom of the DWST and branches into the following.

1. A 6" hotw[illegible] rej ction line (i.e., flow from the condenser hotwell using the condensate pump(s)).
2. A 6" combined AFP suction
3. A 10" hotwell makeup line
4. A 3" water treatment system line

The DWST has a capacity of 100,000 gallons and Technical Specification requires a minimum of 50,000 gallons. Following the loss of AC power, the DWST can be filled from the following sources:

1. The PWST via the PWTP(s)
2. The Recycle PWST via the Recycle PWTP(s)
3. The WT system

The PWST has a capacity of 100,000 gallons and Technical Specification requires a minimum of 80,000 gallons. The DWST is filled from the PWST by the Primary Water Transfer Pumps (PWTP). Each of the 2 PWTP can deliver 200 gpm at 180 psig.

Although there are no technical specifications requiring the availability and/or operability of the Recycle PWST and the Water Treatment System (WT), these are additional sources of water for the steam generators. Since these systems are not included on the "minimum systems" list, the staff gave no credit for their water supplies.

Redundancy

The 2 PWTPs are both powered from the same source, MCC 8-1. Since the PWTP is utilized in the filling of the DWST from the PWST, if the MCC 8-1 were lost, the only way for the PWST water to get to the DWST or to the AFP suction) would be by gravity drainage. The staff calculated that the feed flow required when the required Tech Spec minimum DWST and steam generator inventory is expanded is about 69.5 gpm. If the PWTPs are unavailable due to failure of MCC 8-1, then gravity drainage from the PWST to the AFP suction must be at least 69.5 gpm, and the AFP NPSH requirement for this flow (about ______ft from ______________) must be satisfied. The licensee states ______ _____ ______________________________________.

The staff calculated the maximum length of time the plant can stay at hot shutdown using the initial S/G water inventory and the Technical Specification DWST minimum inventory. These calculations show that approximately 10.4 hours of water supply are available before DWST makeup (from the PWST) must be initiated.

The staff also calculated that the total (Technical Specification required) secondary makeup water inventory, 130,000 gallons, is enough to either keep the plant at hot shutdown for 27 hours, or to complete a shutdown to the point of RHR initiation, 350°F, in about 20 hours. These calculations take no credit for the initial S/G inventory, nor any condensate in the hotwell. The component cooldown times previously discussed show that the ADV, steam generator vents, hoggers and AFP turbines can cool the RCS to the point of RHR initiation (350°F) in about 8.3 hours. (If the ADV were not available due to a single failure, then the AFP RVs could be used to provide additional steam flow).

If the plant stayed at hot shutdown for 4 hours after the scram, then initiated RCS cooldown, the components discussed above could cool the RCS to 350°F in an additional 5.4 hours, or a total elapsed time from the SCRAM of 9.4 hours. Therefore, these calculations show that even if the ADV was unavailable, and the RCS stayed at hot shutdown for 4 hours then initiated the cooldown, the RCS could be at the RHR initiation temperature before the DWST water is expended. However, the staff's calculations are only scoping calculations, and the licensee has stated that about 16 hours are required for RCS cooldown to 350°F.

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Both AFP's take a suction from the hotwell makeup and rejection line. Therefore, the plant has the ability to utilize the water inventory in the condenser hotwell for AFP suction. This could be accomplished by starting a condensate pump and opening (throttling) the hotwell rejection valve (LCV 1317B) or its bypass. The hotwell has a capacity of about 43,200 gallons and a normal operating level of about 33,000 gallons. However, the hotwell contents following a loss of AC and subsequent pump trip and reactor scram cannot be estimated since event times and component coastdowns can't be accurately predicted. Therefore, no credit can be given for this inventory. However, it is highly likely that there would be a significant amount of condensate available to supplement the already mentioned supplies (i.e., steam generator inventory, DWST, PWST, Recycle PWST and the WT system)

Location and Operation

The staff evaluated the equipment discussed above with respect to its location and operability during a loss of offsite AC. The table below gives the equipment's location, the places from where it may be operated, and the equipment's power supply.

Residual Heat Removal System

Task: Removal of core decay heat and RCS latent heat to cool the system from 350°F to 140°F.

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EQUIPMENT	LOCATION	OPERATION	POWER SUPPLY
Demineralized Water Storage Tank	Outside, south (true) of containment (adjacent)	- NA -	No electrical power is needed
Primary water Storage Tank	Outside, about 50 ft from containment	- NA -	No electrical power is needed
Primary Water Transfer Pumps	____________ ____________	____________ and ____________ ____________	MCC 8-1 (480 v) (both pumps)
Recycle PWST	Outside, adjacent to PWST	- N/A -	No electrical is needed
Main Condensate Pumps ()		Main Control room and ____________	
Recycle PWTPs	____________ ____________	____________ and ____________ ____________	

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Discussion

The RHR loop is placed in service after the RCS temperature has been reduced to approximately 350°F and the pressure to less than 400 psig. The RHR system then reduces the RCS temperature to 140°F approximately 20 hours after shutdown, and operates continuously to maintain this temperature as long as it is required by maintenance and/or refueling operations.

The RHR loop consists of 2 heat exchangers, 2 RHR pumps, and the associated piping, valves, and instrumentation necessary for operational control. During plant shutdown, coolant is withdrawn from the loop 1 hot leg through a single letdown line, pumped through the tube side of the residual heat exchangers, and then returned to the cold leg of loop 2 through a single discharge line. Decay Heat load is transferred through the RHR cooler to the component cooling systems which is cooled by component cooling water (normal conditions) or service water (emergency conditions). An alarm will sound in the control room if the RHR flow drops to 2200 gpm.

Remotely operated MOVs provide double valve isolation between the suction and discharge ends of the RHR system and the RCS system. Electrical interlocks are associated with the inboard (closest to RCS) valves which prevent valve opening when RCS pressure is above RHR design pressure. Key control switches are provided for the outboard (closest to RHR) valves to prevent their inadvertent actuation.

Normally, component cooling water flows through the shell side of the RHR coolers. However, following a LOCA, the plant Service Water is directed to the RHR shell side. For this evaluation, the Service Water mode of RHR cooling is considered.

When placing the RHR loop in service, the hot reactor coolant must be introduced into the residual heat removal loop gradually, by regulating the remote-manual control bypass valve and observing the flow indication instrumentation. The RHR to RCS return temperature is controlled by throttling the RHR flow out of the heat exchangers, and CW (or service water) is held constant.

The RHR heat exchangers design conditions are 500 psi gage and 400 F for the tubes and 150 psi gage and 200 F for the shell. Tubes are welded to the tube sheet to prevent coolant leakage into the component cooling system. U-tube design permits expansion caused by large thermal differences between tube and shell.

The RHR pumps are horizontal, centrifugal units designed for 500 psi gage and 400 F with a rated flow of 2,200 gpm at

300 ft minimum developed head. Pump surfaces in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant material. Leakage of radioactive coolant to the atmosphere is reduced to nearly zero by mechanical seals. Any leakoff from the seals drains to the sump and then is pumped to the waste disposal system.

Redundancy

Each RHR pump is sized for one-half the maximum loop flow requirement, and each RHR cooler is sized for one-half the maximum required heat removal capability. The maximum flow and heat removal requirements are when the RHR system is first placed on line 4 hours following a scram from full power. Since there are 2 RHR pumps and coolers, a single failure does not completely disable the RHR system. The use of 2 units also allows maintenance when the plant is shut down and after core decay heat has diminished.

The RHR HX design parameters are given below

Shell side: flow 1.9×10^6 lbm/hr

T_{in} 95°F

T_{out} 105.5°F

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Tube side: Flow 1.1×10^6 lbm/hr

T_{in} 140°F

T_{out} 121.8°F

Under these conditions, each RHR HX is transferring about 20.2×10^6 Btu/hr to the CCW (or Service Water) system, or a total RCS energy removal rate of about 40.4×10^6 Btu/hr. This power corresponds to the core decay heat at about 10 hours after the scram.

Scoping calculations done by the staff for another PWR RHR system similar to Haddam Neck's show that if the flows remain the same, and the tube inlet temperature was 350°F instead of 140°F, the heat transferred increases by a factor of about 4½. (The shell side temperatures will increase due to the larger amount of energy transferred). Assuming this factor (4½) for Haddam Neck, the total RHR heat transfer capability with a 350°F tube inlet temperature would be about 180×10^6 Btu/hr.

This corresponds to the core decay heat at about 200 seconds after the scram (using ANS 5.1). Therefore, scoping calculations indicate that the Haddam Neck RHR system with an inlet temperature of 350°F can remove about 2.92% core decay heat. Since we estimate that the auxiliary feedwater supply would last about

20 hours when cooling down to 350°F, and 2.92% core decay heat corresponds to about 200 seconds after the scram, the RHR system performance, without any failures, is adequate during the loss of AC.

We performed additional scoping calculations on RHR single failures to determine the most limiting single failure. These calculations were done <u>for a PWR RHR system design similar to Haddam Neck's RHR system</u>.* The calculations are summarized below:

RHR Pumps	RHR Hxs	RHR Tube Inlet Temp	Heat Removal (% of full power)
2	2	140°	0.70%
2	2	350°	3.07%
2	1	350°	2.02%
1	1	350°	1.53%

*The plant used, San Onofre 1, has RHR Hxs cooled by CCW system. Haddam Neck's RHR Hxs, under a loss of AC, would be cooled by service water, whose inlet temperature is constant (unlike the CCW system). Therefore, the heat transfer for the Haddam Neck RHR system could be greater than the San Onofre RHR system (e.g., more than the 4½ factor used).

The most limiting single failure from the standpoint of heat removal capacity is the loss, or unavailability of one RHR HX. This reduces the heat transfer capability of the RHR system from 3.07% of full power to 2.02% of full power. However, 2.02% of full power corresponds to about 650 seconds after the scram, which is well before the estimated time when the auxiliary feedwater is expended. Even if an RHR pump and HX were unavailable, the heat transfer capability of the remaining RHR train is about 1.53% of full power, and this power corresponds to about 40 min after the scram.

Since the RHR system has a single drop line with 2 MOVs, a single failure to open of either valve completely disables the RHR system. If this occurred with the reactor vessel head installed, the RCS would be allowed to heat back up and the steam venting mode of decay heat removal would be used. If this occurred with the vessel head removed, the core could be adequately cooled by keeping it flooded using the CVCS or other systems.

Procedure EOP 3.1-20 Rev. 2 "Loss of Residual Heat Removal System" considers three different plant situations when the RHR system is lost: (a) with cavity full, (b) with vessel head in place, and (c) with vessel head closure studs more than half removed. In the case with the head in place, the heat removal is through the steam generators and plant temperature will go above the cold shutdown values.

Location Operation

The staff evaluated the equipment discussed above with respect to its location and operability during a loss of offsite AC. The table below gives the equipment's location, the places from where it may be operated, and the equipment's power supply.

Service Water System

The service water system (SWS) consists of four pumps which supply water from the Connecticut River to a dual header system in which two parallel full size headers supply both the primary and secondary plants. In the turbine building, each header divides into a primary supply and a secondary supply header. Power operated valves at the beginning of each secondary plant header automatically shut to secure secondary service water if offsite AC power is lost and the SWS is reduced to two pumps supplied by the diesel generators (Reference 2). Similar provision is made for shutting off nonessential supplies to primary plant equipment. Also, remote manual valves can be used to substitute SWS flow for component cooling water system flow to the residual heat removal heat exchangers. Each SWS branch connection to a system heat load in both the primary

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EQUIPMENT	LOCATION	OPERATION	POWER SUPPLY
RHR Pumps and Heat Exchangers			
RCS/RHR MOVs	All 4 are inside containment	Control room and ______ ______________	
Cooldown Valves (FCV 602)	Outside containment	Control room and ______ ______________	Control air

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and secondary systems is connected to both headers, and valves permit shut off from either or both headers. During normal operation, both header systems operate in parallel. The SWS pumps and the valves which switch RHR heat exchanger cooling supply from either the SWS or the component cooling water system are controlled from the control room. Power supply for the pumps are from the 480 v emergency buses which can be powered from onsite or offsite sources.

Redundancy

During the shutdown and cooldown following the loss of AC, 2 service water pumps are powered from the emergency power supplies. Upon the loss of all normal alternating current power and after the emergency power supply is established, one service water pump will be started automatically on each diesel generator. If the first service water pump does not start, the power supply is automatically transferred to the second pump on that diesel generator bus.

Operation and Instrumentation

The staff evaluated the location of the various equipment discussed in this section, and the relevant instrumentation available to the control room operators. The table below listed the equipment, its location, the places from where it can be operated, and the equipment's power supply.

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EQUIPMENT	LOCATION	OPERATION	POWER SUPPLY
Service Water Pumps	Screen House	Manually started from the control room, automatically started upon loss of AC & subsequent EDG startup, and __ ____ ________ ______________ ________ ________________	P-37-1A 480V Bus 4 P-37-1B 480V Bus 5 P-37-1C 480V Bus 6 P-37-1D 480V Bus 7
SWS Header Isolation Valves	__________ __________	Automatically upon loss of AC and ____________ _______________________	__________ ____________ ____________

Emergency Power and Instrumentation System

Task: Supply a reliable source of AC and DC power to the necessary equipment and provide sufficient instrumentation to permit control of equipment functions.

Discussion

The staff evaluation of the Emergency Power and instrumentation systems, their reliability, operability, and the associated electrical distribution will be evaluated later under several SEP topics

Chemical and Volume Control System

Task: Provide RCS makeup (due to the contraction of the coolant during the cooldown) and borate the RCS to the necessary shutdown margin.

Discussion

During shutdown and cooldown of the RCS, the Chemical and Volume Control System (CVCS) is used to borate the RCS and provide makeup to accommodate for thermal contraction of the RCS coolant. Two 360 gpm centrifugal charging pumps, powered from redundant emergency 480 V buses, and one 30 gpm positive displacement pump are provided by the CVCS. Charging to the RCS can be accomplished via redundant paths through either the normal charging line or the reactor coolant pump seals. Borated water is provided to the pumps from the Boric

Acid Tank or the refueling water storage tank (RWST) via remotely operated valves. These valves fail as is on loss of power and are operable from the control room. Manually operated valves are available in the Primary Auxiliary Building (PAB) to bypass the remotely operated valves.

The 30 gpm metering pump can provide sufficient charging flow for cooldown at a reduced rate (less than 50°F/hr) provided letdown is secured.

Redundancy

The amount of RCS makeup during cooldown (and filling of the pressurizer) from 542°F to 200°F was calculated by the staff to be about 18,900 gal. Therefore, the BAT alone (12,000 gal) does not provide enough RCS makeup for the plant cooldown to 200°F. However, there are numerous other sources of primary grade water available (e.g., VCT, RWST, PWST, and Recycle PWST).

To ensure the pressurizer level can be controlled during the most rapid cooldown, the staff used the cooldown calculations assuming all the steam venting paths discussed in Section 3.2 and a 4 hour cooldown initiation waiting time. This cooldown rate was initially (i.e., at T_{RCS} = 544°F) slightly greater than 100°F/hr, the Technical Specification maximum (during one hour internal). Our calculations show that a cooldown rate of 100°F/hr causes an RCS liquid contraction rate of about 133 gpm. Since the capacity of each charging pump is about 360 gmp, the pressurizer level can be raised by only 1 charging pump during the most rapid cooldown we calculated, and the other charging pump provides a redundant RCS makeup capability.

The H.ddam Neck technical specifications state that 10,000 gal of 8% solution boric acid are required to meet cold shutdown requirements. Thus, the 12,000 gal at 12% Boric Acid Tank is sufficient, alone, to achieve the required shutdown margin. The RWST can also borate the RCS to the required shutdown margin.

Location and Operation

The staff evaluated the equipment discussed above with respect to its location and operability during a loss of offsite A.C. The table below gives the equipment's location, the places from where it may be operated, and the equipment's power supply.

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EQUIPMENT	LOCATION	OPERATION	POWER SUPPLY
Charging Pumps(2) and Metering Pump	Separate cubicles on the 21'6" level of the PAB	Operable from the control room and ____________ ____________	CPA - CPB - Metering Pump -
Volume Control Tank	Above charging pumps on the 35'6" level of the PAB.	Makeup to the RCT is via the makeup control system which is controlled from the control room and ____ ____________	No electrical power power is needed
RWST	SE corner of PAB (just North of containment)	Level instrumentation and makeup control is from the control room. RWST makeup can also be initiated ____________ ____________	Heaters?
BAT	35'6 level of the PAB.	Level instrumentation ____ ____________ and ______, makeup control ____________.	Heaters?
Boric Acid Transfer Pumps (BATP)	Beneath the BAT, 21'6" level of the PAB	Operable from control Room and ____________ ____________	BATP-1____ BATP-2____

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TABLE 3.1 CLASSIFICATION OF SHUTDOWN SYSTEMS HADDAM NECK PLANT

Components/Subsystems	Quality Group R.G. 1.26	Quality Group Plant Design	Seismic R.G. 1.29	Seismic Plant Design	Remarks
Reactor Control and Protection System	NA*	--	Category I	Note 1	*NA - not applicable Note 1: All systems and equipment are considered capable of withstanding the seismic forces corresponding to a ground acceleration of at least 0.03g unless otherwise noted in this table. (Ref. FDSAR section 2.5)
Auxiliary Feed System (AFS)					
Turbine driven pumps (2)	ASME III Class 3		Category I		
Piping and Valves from pump discharge to valves 156-1 thru 156-A and 182	ASME III Class 3		Category I		Note 2: Pressurizer PORV's, HPSI, and Containment Fan Coolers as discussed in Section 3.2, "Auxiliary Feed Pumps," will be added to this table later.
Demin Wtr. Storage Tank	ASME III Class 3		Category I		
Piping and valves to suction of AFS pumps from Demin. Wtr. Storage Tank	ASME III Class 2		Category I		Plant design info. obtained from FDSAR and CYAPCO letter Switzer to Schwencer of June 29, 1977 (ISI)
Piping and valves from and including valves 156-1 thru 156-4, 182 and main feed valves MOV-11, 12, 13, 14 and 135-1 thru 135-4 to steam generators	ASME III Class 2		Category I		

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TABLE 3.1 (Continued)

Components/Subsystems	Quality Group: R.G. 1.26	Quality Group: Plant Design	Seismic: R.G. 1.29	Seismic: Plant Design	Remarks
Main Steam System					
MS Safety valves (16)	ASME III Class 2		Category I		
MS Atmospheric relief (HICV-1201)	ASME III		Category I		
Piping and valves from Steam generator to and including MS isolation valves and valves HICV-1201, PICV-1206A, B and drain and trap isolation valves TV-1212 and TV-1213 and vent isolation valves.	ASME III Class 2		Category I		
Piping and valves (blow-off) from steam generators to and including blow-off valves TV-1312-1 thru 4 and 506, 515, 522 and 529. Piping from valves PICV-1206A, B to aux. feed pumps including valves SV-1216, A, B	ASME III Class 3		Category I		

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TABLE 3.1 (Continued)

Components/Subsystems	Quality Group R.G. 1.26	Quality Group Plant Design	Seismic R.G. 1.29	Seismic Plant Design	Remarks
Service Water System (SWS)					
SWS pumps (4)	ASME III Class 3	?	Category I		
Piping and valves for containment cooling up to and including valves 263 thru 270	ASME III Class 2	?	Category I		
Piping and valves excluding above and up to and including valves 606, 282, and MOV-1, 2, 3, and 4	ASME III Class 3	?	Category I		Tube sides of CCW and spent fuel HX's are ASME VIII,
Chemical and Volume Control System					FDSAR Table 5.2.1-1
Charging pumps	ASME III Class 2	No code	Category I		
Piping (loop 1) letdown via regen. HX and letdown valves to and including letdown isolation valves	ASME III Class 1	ASA B31.1			

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TABLE 3.1 (Continued)

	Quality Group		Seismic		
Components/Subsystems	R.G. 1.26	Plant Design	R.G. 1.29	Plant Design	Remarks
Regenerative Heat Exchangers (3)	ASME III Class 1	ASME VIII Cases 1270N and 1273N	Category I		
Piping loop drain line via cooler to and including valves 1847 and TV-18471	ASME III Class 1	ASA B31.1	Category I		
Piping and valves pump discharge from and including valves 399 and 296 to RCS	ASME III Class 1	ASA B31.1	Category I		
Piping and valves from pump discharge to containment isolation valves 399 and 296	ASME III Class 2	ASA B31.1	Category I		
Piping from pump discharge via reactor coolant pumps and from TV-1847 to seal water HX	ASME III Class 2	ASA B31.1	Category I		

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TABLE 3.1 (Continued)

Components/Subsystems	Quality Group R.G. 1.26	Quality Group Plant Design	Seismic R.G. 1.29	Seismic Plant Design	Remarks
Piping and valves downstream of letdown isolation valves to the VCT including valves 343A, the interface with ACS, TCV113A via the reactor coolant filter and via relief valves 205, 252 and TCV113B	ASME III Class 2	ASA B31.1	Category I		
Volume Control Tank (VCT) connecting piping and valves up to valves 1847 (relief), 332 (relief) FCV 112C, 246, 251 (relief), 324, 255 and 317	ASME III Class 2	ASA B31.1	Non-Category		
Piping and valves from VCT to charging pumps up to and including valves 33A, B, MOV-366, 320, 369, and the RWST via 372	ASME III Class 2	ASA B313.1	Category I		
Piping and valves downstream of TCV 113A via demineralizers to valve 343A, 220, 234 235 and including demineralizers fill and drain valves	ASME III Class 3	ASA B31.1	Non-Category I		

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TABLE 3.1 (Continued)

Components/Subsystems	Quality Group R.G. 1.26	Quality Group Plant Design	Seismic R.G. 1.29	Seismic Plant Design	Remarks
Boric Acid Tank (BAT)	ASME III Class 3	No Code	Category I		Supply Safety Injection System.
BAT connected piping and valves up to and including valves 347 and 348	ASME III Class 3	ASA B31.1	Category I		
Boric Acid Stainer	ASME III Class 3	No Code	Category I		
Drain Cooler Heat Exchanger (Tube side)	ASME III Class 1	ASME VIII Case 1270N	Category I		
(shell side)	ASME III Class 2	?	Category I		
Seal Water Heat Exchange (Tube side)	ASME III Class 2	ASME VIII Case 1270N	Category I		
(shell side)	ASME III Class 3	ASME VIII	Category I		
Non-Regenerative Heat Exchanger (Tube side)	ASME III Class 2	ASME VIII Case 1270N	Category I		
(shell side)	ASME III Class 3	ASME VIII	Category I		

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TABLE 3.1 (Continued)

Components/Subsystems	Quality Group R.G. 1.26	Quality Group Plant Design	Seismic R.G. 1.29	Seismic Plant Design	Remarks
Mixed-bed demineralizers	ASME III Class 3	ASME VIII Case 1270N	Non-Category I		Class 3 portions of CVCS associated with demineralizers are not required for safe shutdown.
Volume control tank	ASME III Class 2	ASME VIII Case 1270N	Category I		
Seal water injection filters	ASME III Class 2	ASME VIII Case 1270N	Category I		
Boric acid filter	ASME III Class 3	ASME VIII	Category I		
Piping and valves from BAT via boric acid pumps and boric acid filter to valves 329, 369, 366, FCV 112C, 391, 342A	ASME III Class C	ASA B31.1	Category I		
<u>Residual Heat Removal (RHR) System</u>					
RHR pumps (2)	ASME III Class 2	W spec. No code	Category I		RHR pumps provide ECCS containment recirculation

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TABLE 3.1 (Continued)

Components/Subsystems	Quality Group R.G. 1.26	Quality Group Plant Design	Seismic R.G. 1.29	Seismic Plant Design	Remarks
RHR heat exchangers					
(Tube side)	ASME III Class 2	ASME VIII Case 1720 N			
(shell side)	ASME III Class 3	ASME VIII			
Piping and valves to RHR pump suction from RWST, containment sump and valve 781	ASME III Class 2	ASA B31.1	Category I		
Piping and valves from RHR pump discharge and via RHR heat exchangers and bypass to RCS (valves 872 A, B, 803), CVCS, RWST, containment spray, recirculation line to RHR pumps, relief to RWST, and charcoal filter spray	ASME III Class 2	ASA B31.1	Category I		
Process Instrumentation and Controls					
Diesel generators	NA	--	Category I	.17g*	*SER by DRL dated July 1, 1971
Diesel fuel oil, lube oil, starting air	ASME III Class 3	?	Category I	.17g*	
DC power supply system	NA	--	Category I		

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TABLE 3.1 (Continued)

Components/Subsystems	Quality Group R.G. 1.26	Quality Group Plant Design	Seismic R.G. 1.29	Seismic Plant Design	Remarks
Distribution lines, switchgear, control boards and motor control centers	NA	--	Category I		

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4.0 SPECIFIC RHR AND OTHER REQUIREMENTS OF BRANCH TECHNICAL POSITION 5-1

BTP 5-1 contains the functional requirements discussed in Section 3.0 and also detailed requirements applied to specific systems or areas of operation. Each of these specific requirements is presented below with a description of the applicable Haddam Neck system or area of operation.

4.1 RHR System Isolation Requirements

The RHR system shall satisfy the isolation requirements listed below.

1. The following shall be provided in the suction side of the RHR system to isolate it from the RCS.

   (a) Isolation shall be provided by at least two power-operated valves in series. The valve positions shall be indicated in the control room.

   (b) The valves shall have independent divers interlocks to prevent the valves from being opened unless the RCS pressure is below the RHR system design pressure. Failure of a power supply shall not cause any valve to change position.

   (c) The valves shall have independent diverse interlocks to protect against one or both valves being open during an RCS increase above the design pressure of the RHR system.

2. One of the following shall be provided on the discharge side of the RHR system to isolate it from the RCS:

   (a) The valves, position indicators, and interlocks described in item 1(a)-(c).

   (b) One or more check valves in series with a normally closed power-operated valve. The power-operated valve position shall be indicated in the control room. If the RHR system discharge line is used for an ECCS function the power-operated valve is to be opened upon receipt of a safety injection

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signal once the reactor coolant pressure has decreased below the ECCS design pressure.

(c) Three check valves in series, or

(d) Two check valves in series, provided that there are design provisions to permit periodic testing of the check valves for leak tightness and the testing is performed at least annually.

Isolation of the RHR system from the RCS pressure on both the suction and RHR discharge legs is provided by two remotely controlled (from the control room) motor operated RHR valves in series. The two suction and discharge valves nearer the RCS are provided with an interlock which prevents their opening unless RCS pressure, as sensed by the four RCS pressure channels, is less than 400 spig. The two valves farther from the RCS are administratively controlled to prevent misoperation by key locked switches on the control room control board. There is no interlock feature which automatically shuts the valves on an increase of RCS pressure above RHR design pressure (500 psig). The RHR suction and discharge valves fail "as is" on loss of power and have position indication in the control room.

The ECCS functions performed by the RHR system are to provide part of the flow path from the low pressure safety injection (LPSI) pumps to the core deluge supply lines and to supply, using the RHR pumps, long-term ECCS recirculation flow. Following a postulated loss-of-coolant accident (LOCA), when 100,000 gallons of water have been injected by the safety injection (SI) systems, the RHR system is

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started and aligned to take suction on the containment sump to recycle and cool the spilled reactor coolant. If the RCS pressure restricts RHR flow to less than 1000 gpm, recirculation is accomplished by providing RHR flow to the charging pumps (Reference 6). The core deluge flowpath joins the RHR discharge line outside of containment; then, inside containment, the flowpath consists of an eight inch diameter line, which branches from the RHR discharge line. This eight inch line splits into two parallel (six inch diameter) lines each of which again branches into two parallel (four inch diameter) lines leading to the reactor vessel. Isolation of the RHR system from RCS pressure via the core deluge lines is provided by a motor-operated valve and a check valve in each of the six inch diameter deluge lines. The position of the motoroperated valves is displayed in the control room. The motor-operated valves open immediately upon receipt of a safety injection (SI) signal.

Based on the above description, the RHR system deviates from these BTP provisions:

(a) The power operated valves in the core deluge lines open on the SI signal before RCS pressure drops below RHR design pressure.

(b) The RHR suction and discharge isolation valves do not have independent, diverse interlocks to prevent opening the valves until RCS pressure is below RHR design pressure.

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(c) The RHR isolation valves have no interlock feature to close them when RCS pressure increases above the RHR design pressure.

The deviation from the BTP for the core deluge flow paths will be evaluated in the SEP integrated assessment of the Haddam Rock Neck plant. The staff has concluded that no immediate action on this deviation is required because of the high degree of reliability of the check valves in these paths. In addition, any delay in opening of the core deluge motor-operated valves would have to be assessed in light of its effects on the Haddam Neck ECCS analysis.

The deviation for lack of isolation valve diverse interlocks is acceptable because in addition to the single interlock pressure signal on the valves closer to the RCS, the other two valves are key-lock type and are under administrative controls to prevent opening prior to the interlock permissive pressure. By procedure, none of these valve is opened unless RCS pressure is below 400 psig.

The deviation for lack of automatic suction valve closure on increasing RCS pressure is acceptable because, in addition to the administrative and procedural controls on these valves, an alarm is provided at 400 psig to warn the operator that RCS pressure is increasing towards RHR design pressure whenever the Overpressure Protection System (OPS) is enabled. Upon receipt of an alarm, the control

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room operator would be able to terminate the pressure increase or to perform the required procedural steps to isolate the RHR. (See the following discussion of BTP provision C.1, "Pressure Relief Requirements" for information on the OPS.)

4.2 "C. Pressure Relief Requirements

The RHR system shall satisfy the pressure relief requirements listed below.

1. To protect the RHR system against accidental overpressurization when it is in operation (not isolated from the RCS), pressure relief in the RHR system shall be provided with relieving capacity in accordance with the ASME Boiler and Pressure Vessel Code. The most limiting pressure transient during the plant operating condition when the RHR system is not isolated from the RCS shall be considered when selecting the pressure relieving capacity of the RHR system. For example, during shutdown cooling in a PWR with no steam bubble in the pressurizer, inadvertent operation of an additional charging pump or inadvertent opening of an ECCS accumulator valve should be considered in selection of design bases."

The RHR relief valve has a setpoint of 500 psig and a relief capacity of 960 gpm (Reference 11). This relief valve was not sized to accommodate the most limiting pressure increase transients which could be postulated to occur during RHR cooling of the RCS. However, the licensee has analyzed these most severe potential pressure increase transients during the NRC generic review of RCS overpressurization events (Reference 7). To prevent and mitigate these transients, the licensee has made several procedural and hardware modifications. The hardware modifications constitute the Overpressure Protection System (OPS). The OPS relieves RCS pressure via spring safety valves connected to

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the pressurizer and is designed to relieve the pressure increase associated with the worst postulated mass and heat input transients (with exception of the high pressure safety injection pump mass input case for which the licensee has implemented administrative controls) assuming a water-solid RCS and the most limiting single active failure. When the RHR system is cooling the RCS, the OPS also provides overpressure protection for the RHR system.

The staff has evaluated the effects of the worst case mass and heat input events to establish the capability of the OPS and RHR relief to prevent RHR overpressurization. For the mass input case presented in Reference 7, the combined OPS and RHR reliefs prevent RHR pressure from exceeding 550 psig which is 110% of RHR design pressure and is acceptable.

For the heat input case, the data supplied and referenced (Reference 8) by the licensee were extrapolated to include a 50°F steam generator to RCS temperature difference at an RCS temperature of 300°F since the data in Reference 8 only applied to heat input transients associated with RCS temperatures from 180°F to 250°F. Three hundred degrees Fahrenheit was chosen because this is the temperature at which the licensee initiates RHR cooling.

The staff determined that pressure transients which would result from heat input would not exceed 110% of RHR design pressure if both OPS

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reliefs and the RHR relief functioned. Also, it was determined that RHR overpressure would not occur at an RCS temperature of 200°F even assuming the single failure of one of the three valves. The staff then considered the potential for initiating a heat input transient at Haddam Neck when RCS temperature is between 250°F and 300°F. For a heat input transient to occur, the heat from the steam generators must be rapidly transferred to a cooler RCS in a water-solid condition. The means of rapid heat transfer is forced convection caused by a reactor coolant pump start. In its review of overpressurization transients, the staff considered steam generator to RCS temperature differences in excess of 50°F to be unlikely occurrences. The administrative measures proposed by the licensee to reduce the probability of heat impact transients were to 1) minimize plant operation in water-solid conditions, 2) maintain no greater than a 50°F steam generator to RCS temperature difference, and 3) prohibit reactor coolant pump start when steam generator temperature is above RCS temperature. Although items 1 and 2 above would not necessarily preclude a heat addition event, item 3 would. Also, the staff examined the potential for initiating a heat input event during plant cooldown which is the time that steam generator temperature may exceed RCS temperature with RCS temperature above 250°F. The licensee initiates RHR cooling at 300°F after cooling down to that point with the steam generators. Continuing the cooldown with the RHR system and with reactor coolant pumps secured would result in the 50°F difference being fully developed at an RCS temperature of 250°F. As noted before, a heat input event at this temperature would not result in RHR overpressurization even with an assumed single failure.

Based on the above discussion, we concluded that the OPS and RHR relief provide sufficient RHR overpressure protection for RCS temperatures of 250°F or less and that the licensee's procedures acceptably minimize the likelihood of a heat addition overpressure transient at an RCS temperature above 250°F. Therefore, the OPS and RHR relief meet the pressure relief requirements of the BTP.

At present the licensee has proposed to enable the OPS prior to RHR initiation by procedure, and has also proposed making this OPS/RHR sequencing part of the facility technical specifications. The Haddam Neck OPS electrical review is currently underway within NRC.

4.3 "2. Fluid discharged through the RHR system pressure relief valves must be collected and contained such that a stuck open relief valve will not:

(a) Result in flooding of any safety-related equipment.

(b) Reduce the capability of the ECCS below that needed to mitigate the consequences of a postulated LOCA.

(c) Result in a non-isolatable situation in which the water provided to the RCS to maintain the core in a safe condition is discharged outside of the containment."

Fluid discharged through the RHR relief valve is directed to the Refueling Water Storage Tank (RWST) and cannot cause the flooding of any safety-related equipment. Since this valve is connected to the discharge of the RHR pumps, if the valve should stick open core deluge flow and post-LOCA recirculation flow would be affected.

The RHR relief valve flowrate is 960 gpm at a relieving pressure of 500 psig. In a post-LOCA scenario, the pressure felt at the RHR relief will be either LPSI pump discharge pressure, if core deluge flow is being delivered, or RHR pump discharge pressure, if recirculation is in progress. As noted in Reference 9, only one LPSI pump, with a flowrate of 5000 gpm at a discharge pressure of 252 psig, and only one RHR pump, with a flowrate of 2200 gpm at a discharge pressure of 130 psig, are required to be operable to supply 100% of core deluge and recirculation requirements.

The Haddam Neck ECCS are provided with two LPSI and two RHR pumps to meet the redundancy requirements posed by the single failure criterion of the ECCS Interim Acceptance Criteria. Therefore, the leakage through the RHR relief (less than 960 gpm) would result in a lesser reduction in ECCS flow than the loss of a LPSI pump (5000 gpm) or an RHR pump (2200 gpm) which losses have been postulated as single failures in the ECCS analysis.

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The technical specification requirement for ECCS pump operability at Haddam Neck specify that one train of ECCS must be operable whenever the reactor is critical. (One train of ECCS includes one RHR pump and one LPSI pump.) However, no technical specification requirement exists to govern the allowed outage time of the other ECCS train. Such a requirement should exist to maintain the ECCS redundancy assumed in the ECCS analysis. This technical specification requirement, as well as all of the Haddam Neck technical specfications will be reevaluated under SEP Topic XVI, "Technical Specifications."

The fluid discharged through the RHR relief valve goes to the RWST and is still available for RCS cooling via the high and low pressure safety injection systems. However, during post-LOCA recirculation, the fluid may be radioactively contaminated, and leakage of this fluid to the RWST would not be acceptable. Isolation valves in the RHR relief piping are installed with one isolation valve at the relief valve inlet and one in the relief valve discharge line outdoors near the RWST. These valves are both manually operated valves. The isolation valve near the RWST (SI-V-878) can be shut to isolate RHR recirculation flow leakage through the RHR relief valve.

4.4 "3. If interlocks are provided to automatically close the isolation valves when the RCS pressure exceeds the RHR system design pressure, adequate relief capacity shall be provided during the time period while the valves are closing."

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As noted above, these interlocks are not provided. However, the overpressure protection afforded by the RHR relief valve in conjunction with the OPS would provide adequate relief capacity to prevent RCS pressure from exceeding RHR design pressure.

4.5 "D. Pump Protection Requirements

> The design and operating procedures of any RHR system shall have provisions to prevent damage to the RHR system pumps due to overheating, cavitation or loss of adequate pump suction fluid."

The features of the Haddam Neck plant designed to prevent damage to the pumps are provision for pump cooling, a flow recirculation line, and a low flow alarm. Also, indications are available in the control room for RHR flow and valve positions of all remotely operated RHR valves. Either the Service Water System or the Component Cooling Water System can be aligned to provide cooling water to the RHR pump bearings and lubricating oil cooler to help prevent pump overheating. An alarm is provided to alert the operator to a high temperature condition in the pump bearings. In addition, a 3/4-inch line permits the recirculation of some RHR pump flow from the discharge side to the suction side of the pump to prevent overheating caused by no-flow pump operation. An RHR system low flow alarm alerts the operator if a low flow condition occurs with an RHR pump in operation. The availability of adequate net positive suction head (NPSH) will be evaluated during the SEP review of Topic VI-7.E, "ECCS Sump Design and Test for Recirculation Mode Effectiveness."

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4.6 "E. Test Requirements

The isolation valve operability and interlock circuits must be designed so as to permit on line testing when operating in the RHR mode. Testability shall meet the requirements or IEEE Standard 338 and Regulatory Guide 1.22.

The preoperational and initial startup test program shall be in conformance with Regulatory Guide 1.68. The programs for PWRs shall include tests with supporting analysis to (1) confirm the adequate mixing of borated water added prior to or during cooldown can be achieved under natural circulation conditions and permit estimation of the times required to achieve such mixing, and (2) confirm that the cooldown under natural circulation conditions can be achieved within the limits specified in the emergency operating procedures. Comparison with performance of previously tested plants of similar design may be substituted for these tests."

The RHR isolation valve operability and interlocks cannot be tested during the RHR cooling mode of operation. This test requirement is not applicable to the Haddam Neck facility since the installed interlocks function only when the RHR isolation valves are shut.

Regulatory Guide 1.68 was not in existence when the Haddam Neck preoperational and initial startup testing was accomplished. However, a (natural circulation) test was performed to confirm that cooldown under natural circulation conditions is possible (Reference 10). The test involved timing the transit of a cold "slug" of RCS water as it flowed, under natural circulation, around one of the Haddam Neck's four RCS coolant loops. The test results indicated that a natural circulation loop flow of approximately 3% of design loop flow could be achieved about one hour after reactor

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shutdown. This test and subsequent observations of natural circulation flow demonstrate that adequate flow for core cooling exists.

No testing has been performed at Haddam Neck to determine the adequaty of boron mixing under natural circulation flow conditions. However, the staff believes that, with the boric acid concentrations used for shutdown, adequate boron mixing will occur under natural circulation flow.

4.7 "F. Operational Procedures

> The operational procedures for bringing the plant from normal operating power to cold shutdown shall be in conformance with Regulatory Guide 1.33. For pressurized water reactors, the operational procedures shall include specific procedures and information required for cooldown under natural circulation conditions."

Operational procedures reviewed in this comparison of the Haddam Neck to BTP RSB 5-1 are discussed in Section 2.0. All of the procedures required the use of nonsafety-grade equipment for portions of the shutdown operation. No procedures exists for shutdown and cooldown using safety-grade equipment only, but a procedure exists for proceeding to cold shutdown conditions from outside the control room. No procedure exists for plant cooldown using the ECCS system as described in the Section 3.2 discussion of the Auxiliary Feed System. The need for procedures using only safety-grade equipment is not identified in Regulatory Guide 1.33 but stems from the provisions of BTP RSB 5-1

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and SEP Topic VII-3. The staff will consider requiring the licensee to develop these procedures during the integrated SEP assessment of the plant. Therefore, we conclude that the procedures for safe shutdown and cooldown are in conformance with Regulatory Guide 1.33. The plant operating procedures also include a procedure for cooldown using natural circulation.

4.8 "G. Auxiliary Feedwater Supply

> The seismic Category I water supply for the auxiliary feedwater system for a PWR shall have sufficient inventory to permit operation at hot shutdown for at least four hours, followed by cooldown to the conditions permitting operation of the RHR system. The inventory needed for cooldown shall be based on the longest cooldown time needed with either only onsite or only offsite power available with an assumed single failure."

For the discussion of this BTP provision, see Section 3.2.

5.0 RESOLUTION OF SEP TOPICS

The SEP topics associated with safe shutdown have been identified in the INTRODUCTION to this assessment. The following is a discussion of how the Haddam Neck Plant meets the safety objectives of these topics.

5.1 Topic V-10.B RHR System Reliability

The safety objective for this topic is to ensure reliable plant shutdown capability using safety-grade equipment subject to the guidelines of SRP 5.4.7 and BTP RSB 5-1. The Haddam Neck systems have been compared with these criteria, and the results of these comparisons are discusssed in Section 3.0 and 4.0 of this assessment. Based on these discussions, we have concluded that the systems fulfill the topic safety objective subject to the resolution of the following in the SEP integrated assessment:

1. The requirement for plant operating procedures to shutdown and cooldown using safety grade systems only.
2. The requirement for a procedure to cooldown using the ECCS following loss of main and auxiliary feed.

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5.2 Topic V-11.A Requirements for Isolation of High and Low Pressure Systems

The safety objective of this topic is to assure adequate measures are taken to protect low pressure system connected to the primary system from being subjected to excessive pressure which could cause failures and in some cases potentially cause a LOCA outside of containment.

This topic is assessed with regard to the isolation requirements of the RHR system from the RCS. As discussed in Sections 4.1 and 4.2, subject to the following item, which will be considered in the SEP integrated assessment, adequate overpressure protection exists for the RHR system:

1. The need for interlocks on the core deluge motor-operate valves to prevent opening until RCS pressure is below RHR design pressure.

5.3 Topic V-11.B RHR Interlock Requirements

The safety objective of this topic is identical to that of Topic V-11.A. The staff conclusion regarding the Haddam Neck valve interlocks, as discussed in Section 4.1, is that adequate interlocks exists with the

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exception of the potential need for interlocks on the core deluge motor-operated valves.

5.4 Topic VII-3 Systems Required For Safe Shutdown

The Safety objectives of this topic are:

1. To assure the design adequacy of the safe shutdown system to (a) initiate automatically the operation of appropriate systems, including the reactivity control systems, such that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences or postulated accidents, and (b) initiate the operation of systems and components required to bring the plant to a safe shutdown.

2. To assure that the required systems and equipment, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown are located at appropriate locations outside the control room and have a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

3. To assure that only safety grade equipment is required for a PWR plant to bring the reactor coolant system from a high pressure condition to a low pressure cooling condition.

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Safety objective 1(a) will be resolved in the SEP Design Basis Event reviews. These reviews will determine the acceptability of the plant response, including automatic initiation of safe shutdown related systems, to various Design Basis Events, i.e., accidents and transients (Reference 12)

Objective 1(b) relates to availability in the control room of the control and instrumentation systems needed to initiate the operation of the safe shutdown systems and assures that the control and instrumentation systems in the control room are capable of following the plant shutdown from its initiation to its conclusion at cold shutdown conditions. The ability of the Haddam Neck Plant to fulfill objective 1(b) is discussed in the preceding sections of this report. Based on these discussions, we conclude that safety objective 1(b) is met by the safe shutdown system at Haddam Neck subject to the findings of related SEP Electrical, Instrumentation, and Control Topic reviews.

Safety objective 2 requires the capability to shutdown to both hot shutdown and cold shutdown conditions using systems, instrumentation, and controls located outside the control room. The Haddam Neck procedure EOP 3.1-42, "Plant Operation Outside Control Room," provides the necessary steps to take the plant to the hot shutdown condition and to proceed from there to the cold shutdown condition. The procedure, which was made effective on July 12, 1978, contains the

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detailed steps required to achieve the cold shutdown condition; however, it does not define the water sources for the auxiliary feedwater pumps. Portable battery-powered instrumentation is provided at the emergency control point in the cable vault penetration area to measure pressurizer pressure and level, reactor coolant temperature and steam generator level; however, the procedure does not contain a program to ensure that the batteries are functional. The various work locations where operator action or attendance is required have been described in the procedure; however, the duty stations of the individual positions of the operating staff have not been defined, therefore, an individual would not know his area of responsibility without further instructions. The procedure does not address the need for emergency communication equipment at the various duty stations.

Local instrumentation is used for the Boric Acid Mix Tank level. Service Water Flow, and Steam Generator Presure.

The emergency control point is located in the lower level of the cable vault. The cable vault area has an automatically initiated CO_2 fire protection system. It appears that the initiation of the CO_2 system may require evacuation of the emergency control point. In this case, the time period during which control point would be unmanned would be very short and would have a negligible impact on the shutdown and cooldown procedure.

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Based on the information provided in Procedure EOP 3.1-42, and obtained during the safe shutdown site visit, we conclude that the Haddam Neck Plant meets safety objective 2 of Topic VIII-3 with the exception of procedural shortcomings regarding maintenance of batteries for portable instruments, assignments of shutdown duties for shift personnel, emergency communication methods, and water sources for the auxiliary feed pumps. The licensee will be requested to modify his procedures to alleviate these shortcomings.

The adequacy of the safety grade classification of safe shutdown systems at Haddam Neck, to show conformance with safety objective 3, will be completed in part under SEP Topic III-1, "Classification of Structures, Components, and Systems (Seismic and Quality)," and in part under the Design Basis Event reviews. Table 3.1 of this report will be used as input to Topic III-1.

5.5 Topic X Auxiliary Feed System (AFS)

The safety objective for this topic is to assure the AFS can provide adequate cooling water for decay heat removal in the event of loss of all main feedwater using the guidelines of SRP 10.4.9 and BTP ASB 10-1.

The Haddam Neck AFS is described in Section 3.2. This system has been compared with SRP 10.4.9 and BTP ASB 10-1 with the following conclusions:

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1. The Haddam Neck Plant including the AFS will be reevaluated during the SEP with regard to internally and externally generated missiles, pipe whip and jet impingement, quality and seismic design requirements, earthquakes, tornadoes, floods, and the failure of nonessential systems.

2. The AFS conforms to General Design Criteria (GDC) 19, "Control Room," GDC 45, "Inspection of Cooling Water Systems," 46, "Testing of Cooling Water Systems" and Regulatory Guide 1.62, "Manual Initiation of Protection Actions." GDC 5, "Sharing of Structures, Systems, and Components," is not applicable.

3. A passive failure of the common pump suction or discharge headers or the non-essential condensate service line, to which the AFS suction line is attached, would prevent the AFS from supplying feedwater to the steam generators even without an assumed concurrent single active failure. The low probability of a passive failure in the low pressure section line or in the discharge line which is periodically tested under the licensee's inservice inspection program alleviates the need for any immediate corrective measures. The staff intends to examine the need for a long-term improvement in the redundancy of the AFS at Haddam Neck. This will be considered in the SEP integrated assessment of the plant.

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4. Although the AFS does not meet the provision for power diversity of BTP ASB 10-1, the system design does permit emergency feeding of the steam generators with an assumed loss of all AC power; but manual operation of valves in the steam supply lines to the AFS turbines is required. In this case, manual valve operation is permissible because with no feed, the steam generator water inventory can remove decay heat for approximately one hour.

5. The staff is continuing to evaluate feed system waterhammer for the Haddam Neck Plant on a generic basis. SEP Topic V-13, "Waterhammer," applies.

6. The AFS is not automatically initiated and the design does not have capability to automatically terminate feedwater flow to a depressurized steam generator and provide flow to the intact steam generator. This is accomplished by the control room operator. The effect of this provision will be assessed in the main steam line break evaluation for Haddam Neck.

7. The technical specifications for the AFS will be reevaluated against current requirements under SEP Topic XVI, "Technical Specifications."

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6.0 REFERENCES

1. CYAPCO letter D Switzer to A. Schwencer dated June 29, 1977 transmitting the Haddam Neck Inservice Inspection Program.

2. Facility Description and Safety Analysis Report for the Haddam Neck Plant as amended.

3. Supplement to the Safety Evaluation by the Directorate of Reactor Licensing, U.S. Atomic Energy Commission, December 27, 1974.

4. Supporting Information for the Connecticut Yankee Full Term Operating License Application, December 1969.

5. Safety Evaluation by the Directorate of Reactor Licensing, U.S. Atomic Energy Commission, July 1, 1971.

6. CYAPCO letter D. Switzer to R. Purple dated September 29, 1975.

7. Specific Plant Report, Low Temperature RCS Overpressure Protection for Connecticut Yankee, August 1977, transmitted by CYAPCO letter dated September 7, 1977.

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8. Westnghouse Analyses of Overpressure Transients July 1977.

9. Technical Specifications, Appendix A to Facility Operating License DPR-61 for the Haddam Neck Plant.

10. CYAPCO Report "Reactor and Plant Performance Engineering Tests and Tests and Measurements," October 1969, appended to AEC Division of Compliance letter dated December 1969.

11. CYAPCO letter, D. Switzer to A. Schwencer, dated March 1, 1977.

12. Systematic Evaluation Program, Status Summary Report, NUREG-0485.

13. CYAPCO letter, D. Surtzer to K. Goller, dated February 5, 1975 transmitting report "Effects of a High Energy Piping System Break Outside of Containment."

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