September 10, 1982

Docket No. 50-155
LS05-82-09-033

Mr. David J. VandeWalle
Nuclear Licensing Administrator
Consumers Power Company
1945 West Parnall Road
Jackson, Michigan 49201

Dear Mr. VandeWalle:

SUBJECT: BIG ROCK POINT - SEP TOPICS V-10.B, RHR RELIABILITY, V-11.B, RHR INTERLOCK REQUIREMENTS AND VII-3, SYSTEMS REQUIRED FOR SAFE SHUTDOWN

(SAFE SHUTDOWN SYSTEMS REPORT)

A draft evaluation of Safe Shutdown Systems for the Big Rock Point plant was transmitted to you by letter dated May 13, 1981. Your letter of March 30, 1982, provided extensive comments on this evaluation.

Enclosed is the final evaluation of these topics. The staff concludes that, subject to completion of other related SEP topics (e.g., Topic III-1) and other staff programs (e.g., Appendix R), the safe shutdown capability for Big Rock Point is acceptable.

This evaluation will be a basic input to the integrated assessment for your facility. The topic assessments may be revised in the future if your facility design is changed or if NRC criteria relating to this topic are modified before the integrated assessment is completed.

SE04

Sincerely,

DSU USE(18)

ADD: R. Scholl

Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
Division of Licensing

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Enclosure: As stated

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*See previous yellow for additional concurrences.

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Docket No. 50-155
LS05-82

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Nuclear Licensing Administrator
Consumers Power Company
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Dennis M. Crutchfield, Chief
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SEP REVIEW

OF

SAFE SHUTDOWN SYSTEMS

FOR THE

BIG ROCK POINT PLANT

REVISION 1

(August 1982)

<u>TABLE OF CONTENTS</u>

1.0 INTRODUCTION

The Systematic Evaluation Program (SEP) review of the "safe shutdown" subject emcompassed all or parts of the following SEP topics, which are among those identified in the November 25, 1977 NRC Office of Nuclear Reactor 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)

The review was primarily performed during an onsite visit by a team of SEP personnel. This onsite effort, which was performed during the period June 10-12, 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 shutdown condition (reactor shutdown in accordance with technical specifications, temperature above 212°F) and for proceeding to a cold shutdown condition (reactor shutdown in accordance with technical specifications and reactor coolant system at atmospheric pressure). The review for transition from operating to shutdown considered the requirement that the capability exist to perform this evolution from outside the control room. The review was augmented as necessary to assure resolution of the applicable topics, except as noted 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. (No discussion of Topic IX-3 is included in this report. The information gathered during the safe shutdown review will be used to resolve this topic later in the SEP.)

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) System"; and Branch Technical Position RSB 5-1, Rev. 1, "Design Requirements of the Residual Heat Removal System." These documents represent current staff criteria for the review of applications for operating licenses.

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

As noted above, the five 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-3.C (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 Consideration), 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 review of those topics. 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 under other topics in the SEP.

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 SEP 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 Big Rock Point):

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 Big Rock Point Plant.

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 Break 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 Collant Flow) Primary Pump Rotor Seizure) (Primary Pump Shaft Break)	Probability
	VII (Defined above)	Consequences

*For a listing of DBE groups and generic topics see the Systematic Evaluation Program, Status Summary Report, NUREG 0485, April 1982.

Piping System Passive Failures

The NRC staff normally postulates piping system passive failures as 1) accident initiating events in accordance with staff positions on piping failures inside and outside containment, 2) system leaks during long-term coolant recirculation following a LOCA, and 3) failures resulting from hazards such as earthquakes, tornado missiles, etc. In this evaluation, certain piping system passive failures have been assumed beyond those normally postulated by the staff, e.g., the catastrophic failure of moderate energy systems. These assumptions were made to demonstrate safe shutdown system redundancy given the complete failure of these systems and to facilitate future SEP reviews of DBEs and other topics which will use the safe shutdown evaluation as a source of data for the SEP facilities. SRP 5.4.7 and BTP RSB 5-1 do not require the assumption of piping system passive failures.

Credit for Operating Procedures

For the safe shutdown evaluation, the staff may give credit for facility operating procedures as alternate means of meeting regulatory guidelines. Those procedural requirements identified as essential for acceptance of an SEP topic or DBE will be carried through the review process and considered in the integrated assessment of the facility. At that time, we will decide which procedures are so important to acceptance of a topic that an administrative method must be established to ensure that in the future, operating procedures are not changed without appropriate consideration of their importance to the SEP topic evaluations.

2.0 DISCUSSION

2.1 Normal Plant Shutdown and Cooldown

There are two general procedures utilized to perform a shutdown from full power to cold shutdown. The first procedure, entitled "Plant Shutdown to Hot Shutdown," is used to place the plant in a hot shutdown condition. The reactor power is reduced by the sequential insertion of the control rods, and the initial pressure regulator (IPR) is used to reduce the turbine load. At 50 MWe, one reactor feedwater pump is taken out of service; and, at 45 Mwe, one condensate pump is stopped. When the generator load is reduced to zero, the turbine is tripped and all control rods are inserted. The cooldown is continued with the steam jet air ejectors and the main condenser. When the reactor pressure has been reduced to below 200 psig, the shutdown cooling system is placed in service in accordance with System Operating Procedure "Reactor Shutdown System". When condenser vacuum can no longer be maintained, the operator will break condenser vacuum and remove the air ejectors from service. The cooldown is continued in accordance with operating procedure, "Plant Shutdown to Cold Shutdown" with the use of the shutdown cooling system until the reactor system is less than 212°F and at atmospheric pressure.

During the entire shutdown, the control rod drive system and the reactor cleanup system are normally in service. The control rod drive system will be injecting cold water to the reactor and the non-regenerative heat exchanger in the cleanup system will be removing heat from the reactor. These two systems will enhance the cooldown of the reactor.

The reactor shutdown cooling is accomplished by transferring heat from the reactor coolant, through the shutdown heat exchangers, to the reactor cooling water system. The decay heat is, then, transferred from the reactor cooling water system, through the reactor cooling water heat exchangers to the service water system where the waste heat goes to the discharge canal.

The loss of each of these three systems has been addressed in off-normal procedures:

A. "Loss of Shutdown Cooling System"

B. "Loss of Reactor Cooling Water System"

C. "Loss of Service Water"

The "Loss of Shutdown Cooling System" procedure directs the operator to put the standby shutdown cooling train in service; to increase the cold water flow to the reactor by starting the standby control rod drive pump and to increase the flow to the cleanup system to maximum capacity. If these immediate actions are inadequate, the operator is directed to use the core spray system, as required, to control steaming in the reactor vessel.

The "Loss of Reactor Cooling Water System" procedure describes the automatic start of the standby reactor cooling water pump.

The "Loss of Service Water" procedure provides direction to use water from the fire system if the service water pumps fail.

Interlocks are provided on the shutdown cooling system isolation valves that will prevent the opening of these valves if the reactor pressure is greater than 300 psig.

The following information was taken from operating logs and records to document an instance where a cooldown was achieved with the use of only one shutdown cooling pump. The information is from a shutdown initiated on January 16, 1975:

Jan. 16, 1975	1245	Start reducing plant load to take plant off line
	1344	No. 1 Reactor Feedpump off
	2044	Turbine off the line

Jan. 17, 1975	0045	Steam supply off to steam jet air ejectors
	0246	Broke condenser vacuum
	0322	No. 2 reactor shutdown cooling pump on. Reactor system temperature was approximately 300°F.
	0700	Water temperature to the shutdown cooling system was 191°F (<212°F)

2.2 Shutdown and Cooldown with Loss of Offsite Power

The required operations to place the unit in a shutdown condition are given in Off Normal Procedure (ONP) "Loss of Station Power." When a loss of power occurs, a series of automatic operations occur:

A. Reactor scram
B. Turbine trip
C. Emergency condenser outlet valves open
D. Emergency diesel generator start

The operator is then directed to operate the outlet valves on the emergency condenser to control the cooldown rate. Subsequent switching operations are defined in Off-Normal Procedure (ONP) 2.36, "Loss of Station Power," where the essential plant loads are connected to the diesel generator power source. This procedure provides the operator with guidance and instructions to perform a cooldown to cold shutdown conditions. Use of the emergency condenser is included.

The licensee has developed a procedure that considers the loss of the emergency diesel generator during the loss of offsite power: "Loss of Emergency Diesel Generator." A second, portable 250 kw diesel generator is dedicated to the plant and can be placed in service within 24 hours with temporary cables between the portable diesel unit and the input terminals to the breaker (52-2B27) that normally serves the existing diesel generator.

The licensee has developed an Emergency Procedure, "Loss of D.C. Power System," that defines the operator action that is required to take the plant to cold shutdown in the event there is either a sudden or gradual loss of d-c power.

The emergency condenser contains enough water to remove the decay heat for approximately four hours after the reactor is scrammed. The normal makeup to the emergency condenser is provided from the demineralized water system; however, provisions have been made to supply fire water to the emergency condenser if the normal supply fails. The procedure, "Loss of Emergency Condenser Cooling Systems," provides direction for using either the electric or diesel driven fire pump to supply water to the emergency condenser.

The plant experienced a loss of offsite power on January 25, 1972. Records of the loss of offsite power contained the information that the emergency diesel generator was in service for slightly over two hours, all non-essential equipment was removed from service, and required equipment was returned to service on emergency power as required.

3.0 SAFE SHUTDOWN FUNCTIONS AND METHODS

This section will describe the systems available at Big Rock Point to perform the necessary functions for the safe shutdown of the reactor plant following either the loss of offsite power or the loss of onsite AC power. A comparison of these systems with current NRC criteria will be presented in Section 4.0.

The loss of offsite and onsite AC power are not considered to be concurrent or sequential but are considered to be independent occurrences.

A description of how the systems are used for safe shutdown is provided in the following sections.

a) Emergency Condenser - The emergency condenser is a natural circulation cooling device which condenses steam from the steam drum and returns the cooled condensate. The emergency condenser is activated automatically when reactor pressure reaches 100 psi above normal reactor pressure. The primary side has two tube bundles, each with its own inlet line and valve and its own outlet line and valve. The inlet valves are AC operated and supplied from non-emergency bus 2A and are normally open during plant operation. The outlet valves are DC operated and are supplied from a common bus, which is normally powered by the 125 VDC station battery. The bus can also be energized by either the No. 1 or No. 1A station battery charger. On loss of AC power, the outlet valves open automatically. The shell side normally contains sufficient water to dissipate decay heat for 4 hours following reactor scram. Make up is automatic through an air operated valve from the demineralized water system when low level is reached. The air operated makeup valve fails closed on loss of air supply. A back-up flow path and water supply is provided via a local manual valve inside containment connected to the fire water system. This provides an unlimited source of water to the shell side of the emergency condenser on loss of air and loss of AC power, although operator action inside containment is needed. A modification is planned that will allow control room operation of this valve. Emergency condenser shell level is monitored by a level instrument and a level switch with a low level alarm.

b) Shutdown Heat Removal System (Shutdown Cooling System) - The shutdown heat removal system takes suction on the reactor vessel and returns to one of the recirculation loops downstream of the recirculation pump. The suction and return lines each have dual series isolation valves which are interlocked to prevent opening at RCS pressures above 300 psig. They also close automatically at 300 psi increasing. Redundant trains of shutdown pumps and heat exchangers are provided. However, pumps and heat exchangers are not interchangeable between trains. Pump and valve control valve operations may be done either in the control room or local manually. Use of the No. 1 SCS train is preferred, but the No. 2 pump and heat exchanger are also available for remote operation from the control room. Flow control operations are performed locally. Cooling water to the shell side of the shutdown heat exchanger is provided from the closed loop reactor cooling water system which is, in turn, cooled by the service water system. Experience has shown that the heat removal capability of the shutdown cooling system is sufficient to cool the reactor using a single pump and heat exchanger.

Reactor shutdown pump power is supplied by 480 V Bus 1A and 2A which are de-energized on loss of offsite power. Reactor cooling water pumps are powered from the same busses. It is possible to manually align electrical power busses to provide power to the shutdown cooling and reactor cooling water pumps from the emergency diesel. It is also possible to manually valve in coolant from the fire protection system to the service water system. These steps make it possible to provide a path for heat removal to the final heat sink via the shutdown cooling system, the reactor cooling water system, and the service water system in the case of loss of offsite power.

The SCS isolation valves are closed and their breakers (located inside containment) opened whenever primary system pressure is greater than or equal to 300 psig. This requirement is embodied in the operating procedure for the SCS as well as in the Technical Specifications. Therefore, manual action inside containment would be necessary to use the SCS for plant shutdown.

c) Core Spray Systems (CSS) - Two separate low pressure core spray systems are available to inject low pressure water into the core. Each injection path contains two normally closed motor operated valves in series. The valves in the lines feeding the core spray ring are DC powered from the station battery and charger. The valves in the lines feeding the redundant core spray nozzle are AC powered from the emergency diesel. Thus, loss of offsite power and a single failure will not disable both systems.

The water supply to each core spray is from the fire water system. Use of the core spray system as a method for safe shutdown requires a letdown path to allow continued injection of cooling water. The blowdown line through the reactor water cleanup system (RWCS) is one option but loss of air supply would disable this path. Power to an air compressor is restored manually by procedure immediately following loss of offsite power. However, the capacity of the cleanup system may not be sufficient. The RDS described below provides a more reliable blowdown capability.

d) Reactor Depressurization System (RDS) - The reactor depressurization system is a fully qualified safety system to automatically depressurize the reactor to allow use of the low pressure core spray system to cool the core. It can also be operated in a manual mode. The system consists of four parallel blowdown paths connected to the main steam header. Each path has an air operated isolation valve which fails open on loss of air and a DC powered valve that fails closed on loss of DC power. DC power to each of these valves is supplied by a separate battery pack to assure operation of 3 of 4 valves in the case of a single failure.

Use of the RDS as a safe shutdown system would be a last choice method because it discharges primary coolant to the containment atmosphere and would require extensive cleanup.

e) Fire Water System (FWS) - The fire system is a single pipe run fed by an electric fire pump and a diesel fire pump in the crib house. The electric fire pump is powered from the emergency diesel generator. The diesel fire pump has its own starting battery and provides a completely independent supply. Thus, loss of offsite power and a single failure will not disable this source of water.

The fire water system supplies water directly to the core spray ring, the redundant core spray nozzle and the core spray heat exchanger, and, by local manual action, through normally closed valves inside containment to the emergency condenser and the service water system. Thus, each of these systems is provided with a source of water in the event of loss of offsite power and the most limiting single failure (provided that the containment is habitable).

As reported in a letter from D. P. Hoffman (Consumers Power Company) to D. M. Crutchfield (NRC), dated October 16, 1981, the licensee has initiated a modification to provide a dc power supply and controls for the emergency condenser valve for operation from the control room.

f) Post Incident Cooling System (PICS) - Long-term cooling is provided by the recirculation mode of the post-incident cooling system following blowdown of the reactor via the RDS system. Following depressurization, the low pressure core spray system would cool the core. The containment begins to fill with condensed steam from the RDS. When the water level in containment reaches approximately the 587' elevation, the valves are realigned for the recirculation mode. Manual action is required to close fire system isolation valves VFP 29 and VFP 30 and to remotely open the fire system feed to the core spray heat exchanger from the control room. A flow path can be established with steam release from the reactor through the RDS valves into containment;

condensing in the sump, then through the strainers, through the core spray pumps and heat exchangers to the core spray ring or redundant core spray nozzle to the reactor core. The shell side of the core spray heat exchanger is cooled by the fire system. The core spray pumps are powered from electrical buses 1A and 2A which must be loaded manually to the emergency diesel generator. Thus, this system, in conjunction with the RDS, provides a means of cooling the reactor for an indefinite time given loss of offsite power and the most limiting single failure.

4.0 COMPARISON OF SHUTDOWN AND COOLDOWN SYSTEM WITH CURRENT NRC CRITERIA

The current criteria used in the evaluation of the design of systems required to achieve cold shutdown for a new facility are listed in the Standard Review Plan (SRP) Section 5.4.7 and Branch Technical Position RSB 5-1 Rev. 1 and Regulatory Guide 1.139, "Guidance for Residual Heat Removal." This section discusses the comparison of these criteria with the safe shutdown systems of the Big Rock Point plant. This comparison will be done by quoting a section of the Branch Technical Position RSB 5-1 and then discussing the degree to which the plant meets the requirements of that particular section.

4.1 Functional Requirements

The systems(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

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*Processes involved in cooldown are heat removal, depressurization, flow circulation, and reactivity control. The cold shutdown conditions, as described in the Standard Technical Specifications, refers to a subcritical reactor with a reactor coolant temperature no greater than 200°F for a PWR and 212°F for a BWR.

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.

Background

A "safety grade" system is defined, in the NUREG 0138* discussion of issue No. 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 for Nuclear Power Plant Protection Systems, (IEEE 279). Big Rock Point was constructed prior to the issuance of Regulatory Guides 1.26 and 1.29 (as Safety Guides 26 and 29 on 3/23/72 and 6/7/72 respectively) and IEEE 279, dated August 30, 1978.

General Design Criteria (GDC) 1 requires that systems 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 be kept.

Regulatory Guide (RG) 1.26 provides the current NRC criteria for quality group classification of safety-related systems. Although RG 1.26 was not in effect when Big Rock Point was constructed, the systems at Big Rock are being classified in accordance with this guide as part of the SEP. In general, the high pressure system at

*Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum from Director, NRR to NRR Staff, NUREG 0138, November 1976.

Big Rock were designed to and built to the 1955 version of ASME Boiler and Pressure Vessel Code, Section 1, Nuclear Code Cases 1270N and 1273N, and ASA B31.1, as shown in Table 4.1. Although the safety-related systems at Big Rock were not designed, fabricated, erected, and tested using RG 1.26, the maintenance and modification of systems is currently conducted in accordance with this guide. For example, the RDS was designed and built to the standards of those regulatory guides.

At the time the Big Rock Point was licensed, the NRC criteria for QA were not developed. The QA program for operation of Big Rock, SEP Topic XVII, was approved by the staff on September 17, 1976.

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.

The effects of tornadoes will be reevaluated during the course of the SEP in Topics II-A "Severe Weather Phenomena," III-2 "Wind and Tornado-loadings," and III-4.A "Tornado Missiles."

Floods and flood effects will be reassessed in the SEP review under Topics II-3.B "Flooding Potential and Protection Requirements," and III-3 "Hydrodynamic Loads."

Within the SEP review, the potential for and consequences of a seismic event at the Big Rock Point site 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.

TABLE 4.1 CLASSIFICATION OF SAFE SHUTDOWN SYSTEMS BIG ROCK POINT

Components/Subsystems	Quality Group: R.G. 1.26	Quality Group: Plant Design	Seismic: R.G. 1.29	Seismic: Plant Design	Remarks
Emergency Condenser System					
Piping, from steam drum up to and including MOV's 7052, 7062	ASME III Class 1	ASA B 31.1 (1955) ASME	I	0.05g	
Condenser overflow and drain line	ASME III Class 2	ASA B 31.1	I	0.05g	
Piping, from steam drum up to and including MOV's 7053, 7063	ASME III Class 1	ASA B 31.1 (1955) ASME	I	0.05g	
Remaining main system piping to and from condenser	ASME III Class 1	?	I	0.05g	
Emergency condenser shell	ASME III Class 2	ASME VIII with code cases 1270N, 1272N	I	0.05g	
Emergency condenser, tube bundles	ASME III Class 1	ASME VIII Code case 1270N	I	0.05g	
Cooling water feed lines to condenser shell from fire system and demineralized water storage	ASME III Class 3	?	I	0.025/0.050[1]	
Condenser vent line up to and including exterior containment isolation valve	ASME III Class 2	?	I	0.025/0.050[1]	

TABLE 4.1 (Continued)

Components/Subsystems	Quality Group R.G. 1.26	Quality Group Plant Design	Seismic R.G. 1.29	Seismic Plant Design	Remarks
<u>Core Spray and Post Incident Cooling System</u>					
Pumps	ASME III Class 2	Manufacturer	I	UBC Zone 1 1958	
Heat Exchanger (Tube side)	ASME III Class 2	ASME Sec. VIII Code Case 1272N TEMA Class R	?	UBC Zone 1 1958	
(Shell side)	ASME III Class 3	ASME Sec. VIII TEMA Class R	?	UBC Zone 1 1958	
Ring Sparger	ASME III Class 2	ASME III	I	0.05g	
Piping, from reactor vessel up to and including MOV 7061	ASME III Class 2	ASA B 31.1 (1955)	I	0.05g	
Piping, from MOV 7061 up to and including check valve	ASME III Class 2	ASA B 31.1 (1955)	I	0.05g	
Piping and valves added to system when backup core spray	ASME III Class 2	USAS B.31.1 USAS B 31.7 ASME Draft code for pumps and valves	I	0.05g	
Core Spray Nozzle	ASME III Class 2	USAS B.31.1 USAS B 31.7 ASME Draft code for pumps and valves	?	?	
All other piping and valves	ASME III Class 2	ASA B 31.1 (1955)	I	UBC Zone 1 1958	

Table 4.1 (Continued)

Components/Subsystems	Quality Group: R.G. 1.26	Quality Group: Plant Design	Seismic: R.G. 1.29	Seismic: Plant Design	Remarks
Reactor Depressurization System					
Piping and Valves from steam drum to RDS valves	ASME III Class 1	ASME III	I	I	
RDS (relief) valves	ASME III Class 1	ASME III Sub. Sec. NB Class 1	I	I	
Discharge lines	ASME III Class 2	ANSI B 31.1 (1973)	I	I	
Fire Protection System					
Fire Pumps	ASME III Class 2	Manufacturer	I	UBC Zone 1 1958	
Piping, Valves, up to Enclosure spray lines including core spray heat exchanger	ASME III Class 2	ASA B 31.1 (1955)	I	0.025g	
Discharge line to drainage ditch from core spray heat exchanger	ASME III Class 3		I	0.025g	
Emergency Power System					
DC power supply system	NA	-	I	UBC Zone 1 1958	

Table 4.1 (Continued)

Components/Subsystems	Quality Group R.G. 1.26	Quality Group Plant Design	Seismic R.G. 1.29	Seismic Plant Design	Remarks
Emergency Power System (Continued)					
Diesel generator	NA	Diesel DEM of Stds Exciter & Generator (AIEE, ASA, NEMA)	I	UBC Zone 1 1958	
Distribution lines, switchgear control boards, motor control center	NA	-	I	?	
Diesel generator mechanical auxiliaries	ASME III Class C	?	I	?	
Safe Shutdown Instrumentation and Control	NA	-	I	?	

Table 4.1 Note (1):

The Final Hazards Summary Report states that the reactor enclosure and equipment within is designed to withstand ground acceleration equivalent to 0.05g; equipment and structures outside are designed to withstand a ground acceleration of 0.025g.

The Big Rock Point fire protection reevaluation resulting from the Browns Ferry fire is currently underway in the NRC staff. The results of this reevaluation will be integrated into the SEP assessment of Big Rock Point.

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 reevaluate the various aspects of this criterion when reviewing topics III-12 "Environmental Qualification of Safety-Related Equipment" (USI A-24), III-5.A "Effects of Pipe Breaks Inside Containment," III-5.B "Pipe Breaks Outside Containment," and III-4 "Missile Generation and Protection."

GDC 5 is not applicable for the Big Rock Point because it does not share any equipment with other power units.

Although other systems are available to perform shutdown and cooldown functions as described in Section 3.0, based on our review of systems available at Big Rock Point to accomplish these functions in accordance with the provisions of BTP RSB 5-1, we have determined that the following minimum number of systems is required:

1. Reactor Protection and Trip System (No discussion included)
2. Emergency Condenser
3. Fire Protection Water System
4. Reactor Depressurization System
5. Core Spray Systems
6. Post Incident Cooling System
7. Instrumentation for Shutdown and Cooldown*
8. Emergency Power (AC and DC) for the Above Systems and Equipment

*Safe shutdown instruments are identified in Table 4.2.

TABLE 4.2 LIST OF SAFE SHUTDOWN INSTRUMENTS

Component/System	Instrument	Instrument Location	Reference
Reactor System	Steam drum level (LE-1D25 A&B, LI-1A77 and 1D59, LT-1A18, 1D13)	LE - Inside Containment LI - Control Room LT - Inside Containment	DWG M-121
	Steam drum pressure (PT-1A07, PI-1A49, PR-1A09)	PT -Inside Containment PI, PR - Control Room	DWG M-121
Emergency Condenser	Shell level (LT-3150, LI-3305 and LS-3549)	LT - Inside Containment LI - Control Room LS - Inside Containment	DWG M-107
Fire Water System	Fire System pressure (PI-338)	PI - Screenhouse	DWG M-123
Core Spray System	CS flow (FT-2162, FI-2335)	FT - Inside Containment FI - Control Room	DWG M-123
Backup Core Spray	CS flow (FT-2163, FI-2336)	FT - Inside Containment FI - Control Room	DWG M-123
Core Spray Recirculation System	CS Recirc press (PS-638)	PS - Locally Mounted Alarm - Control Room	DWG M-123
	Containment water level (LS-3562 thru 3565)	LS - Inside Containment Alarm - Control Room	DWG M-123
Emergency AC Power	Emergency Diesel voltage and current indication	Control Room	DWG 0740G30101
Emergency DC Power	125V DC System voltage indication	Control Room	DWG 0740G30102

Table 4.1 lists these safe shutdown systems along with a comparison of present design criteria with the criteria to which these systems were designed. Table 4.3 provides the power supply and locations of these systems.

The instrumentation listed in Table 4.2 represents those parameters that indicate overall reactor performance (e.g., steam drum level, pressure) and those instruments that monitor performance of the systems being used for the shutdown (e.g., emergency condenser level). The latter set is included to enable the operator to detect degradation in system performance prior to loss of function.

Some of the instrumentation listed would not normally be needed for a shutdown. If the emergency condenser is available, only steam drum level, steam drum or reactor pressure and emergency condenser shell level would be needed. As stated in a letter from R. Vincent to D. Crutchfield of March 30, 1982, at least two control room indications of each of these parameters is provided. Additional readouts will be provided at the local shutdown panel (when installed).

If the emergency condenser cannot be used, other instrumentation would be used to monitor RDS, PICS performance, such as containment water level. It would also be desirable to have flow indications for the post-incident cooling system.

The emergency condenser provides the most desirable means of decay heat removal in those situations in which the main condenser is not available for cooldown. The tube side of the condenser is designed for primary system pressure. Redundant inlet and outlet flow paths are available. However, the outlet valves are powered by a common DC bus and would not meet the requirements of IEEE 279 for single failure and separation. Therefore, with an assumed loss of offsite power (shutdown with only onsite power) and a single failure which disables the station 125 VDC bus, the emergency condenser DC outlet valves would be inoperable and the emergency condenser could not be used for shutdown. In this case, the RDS, core spray system, and post incident cooling system are operable and provide an acceptable means to depressurize and cool the reactor. Depressurization of the reactor with RDS, coolant injection with the core spray systems, and long term cooling by the post-incident cooling system provide this ability. However, because the RDS discharges to containment, and its use would require an extensive containment cleanup effort, this is not the most desirable cooldown method.

Activation of the RDS and core spray for shutdown with loss of offsite power and an assumed single failure can be done from the control room. However, realignment of the post-incident cooling system for long term cooling requires operator action outside the control room but not inside containment.

Activation of RDS results in a very rapid cooldown. Blowdown with RDS is rapid and the coolant temperature follows at saturation conditions. This is followed by injection of cool water from the core spray (fire water) system and then recirculation using the post incident cooling system core spray heat exchanger.

If DC power is not lost the emergency condenser is used for cooldown. Experience at the plant has shown that the heat removal capacity of the emergency condenser is large enough that it is necessary to take action to limit the cooldown to within Technical Specification limits. Plant experience has also shown that the emergency condenser and a single shutdown cooling system pump and heat exchanger are sufficient to cool the plant to cold shutdown within 36 hours. See Appendix A for an evaluation of the capability of the plant systems to perform this cooldown.

Although the Shutdown Cooling System is normally used to attain cold shutdown conditions during routine shutdown of the plant, it is susceptible to a failure to open of either a single suction or discharge isolation valve located inside the containment sphere.

TABLE 4.3 SAFE SHUTDOWN SYSTEM POWER SUPPLY AND LOCATION

Component/System	Power Supply	Location
Fire System Pumps*	Diesel driven - diesel engine Motor driven - 480 V MCC Bus 2B	Screen House (583')
Core Spray Recirculation Pumps 1 and 2	#1 - 480 V MCC Bus 1A #2 - 480 V MCC Bus 2A	Unloading Dock (583')
480 V MCC Bus 1A	Offsite power or Emergency Diesel	Electrical Equipment Room (591')
480 V MCC Bus 2A	Offsite power or Emergency Diesel	Electrical Equipment Room (591')
480 V MCC Bus 2B	Emergency Diesel or Bus 2A	Electrical Equipment Room (591')
Emergency Diesel Generator	-	Screen House (589')
125 V Battery	-	Electrical Equipment Room (591')

*The fire system pumps are also the core spray and backup core spray pumps.

Furthermore, operator entry to containment is necessary to restore power to the valve breakers for remote valve operation. The isolation valves are equipped with handwheels for manual operation in the event of an electrical malfunction. However, the RDS, core spray, and post-incident cooling systems can be used to attain cold shutdown, if required; and these systems are not susceptible to single failures.

4.2 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 diverse 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.

    Evaluation

    At Big Rock Point, the RHR system is called the shutdown cooling system (SCS) and is located inside the containment sphere. Isolation is provided by two power-operated valves in series with indication in the control room. The SCS isolation valves are interlocked to prevent opening at RCS pressure above 300 psig which is the SCS design pressure. Either pressure switch will control both the inboard and outboard isolation valves. The valves fail as is on loss of power.

The pressure sensors described above also provide the signal to close the SCS isolation valves automatically when primary coolant system pressure increases above 300 psig. The closure time is approximately 60 seconds.

As discussed in a letter from R. A. Vincent to D. M. Crutchfield dated November 12, 1981, a single failure of a pressure switch and/or failure of a pressure switch auxiliary relay will not prevent the interlock from keeping one valve closed in the suction line and one in the discharge line.

As previously discussed, the isolation valves are closed, with the breakers (inside containment) opened when primary system pressure is above 300 psig.

Therefore, as discussed in the staff's evaluation transmitted by a letter from D. Crutchfield to D. Vandewalle (LS05-82-08-18) dated August 10, 1982, the staff concludes that modifications to provide diversity or redundancy for the interlocks will not provide a significant improvement in the protection of the public health and safety.

Requirement

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 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.

Evaluation

At Big Rock Point, the provisions for the discharge side are the same as described above for the suction side.

4.3 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 the design bases.

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.

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

Evaluation

At Big Rock Point, two small relief valves set at 300 psig are installed in the SCS. Relief capacity of each valve is approximately 25 gpm. No significant pressure transients are expected because BWR pressures are determined by saturated steam conditions.

The relief valve discharge drains to the containment enclosure sump and would not impact safety related equipment.

4.4 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.

Evaluation

The Shutdown Cooling System pumps are tripped only on pump overload or by local manual action. There is no protection from overheating, cavitation or loss of pump suction fluid. However, the deviation from this BTP provisions is acceptable because the facility possesses other means to remove core decay heat which are redundant to the Shutdown Cooling System pumps.

4.5 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 of IEEE Standard 338 and Regulatory Guide 1.22.

Evaluation

The SCS interlock and auto closure setpoints are checked each refueling and the valves are exercised to assure operability. The licensee has stated that the tests meet the intent of Regulatory Guide 1.22.

4.6 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 for cooldown under natural circulation conditions.

Evaluation

Operational procedures reviewed in this comparison of the Big Rock Point plant to BTP RSB 5-1 are discussed in Section 2.0.

We have concluded that the existing procedures for safe shutdown and cooldown are in conformance with Regulatory Guide 1.33.

The procedure for loss of station power provides guidance and instruction for cooldown to cold shutdown, including use of the emergency condenser. In the Subsequent Operator Action section of ONP 2.36 "Loss of Station Power" the operator is instructed to refer to Emergency Procedure EMP 3.3 "Loss of Coolant" during the cooldown to cold shutdown. This latter procedures provides instructions to bring the plant into the long-term cooling mode using the RDS, CSS, PICS and FWS. System operating procedures provide the operator with information on both automatic operation of these systems and manual actuation, including the manual switchover to recirculation mode.

In addition, the licensee is participating in the General Electric Owners Group preparation of BWR operating procedures, which will address the use of safety grade system for shutdown. Therefore, the staff concludes that reliable plant shutdown capability with safety-grade equipment will be assured.

4.7 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 4 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.

Evaluation

Boiling water reactors such as Big Rock Point do not have an auxiliary feed system. However, the cooling water requirements for a safe shutdown of the facility, using the systems identified in Section 4.1 are evaluated in Appendix A.

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 facility 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 Big Rock Point systems have been compared with these criteria, and the results of these comparisons are discussed in Section 4.0 of this assessment. Because it does not contain system redundancy (single letdown and return lines), the Shutdown Cooling System, which performs the function of a Residual Heat Removal System, does not satisfy the review guidelines. However, we have concluded that the other systems at Big Rock Point fulfill the safety objective. The staff notes the following:

1. The redundant emergency condenser condensate valves are powered by a single DC bus and so are susceptible to the single failure of this bus, although several sources are available to energize this bus. This single failure in conjunction with loss of offsite power would require the use of RDS and Core Spray for cooldown. Since an alternate method of shutdown exists, albeit one with undesirable operational consequences, and given the demonstrated low frequency of total loss of offsite power, the possible single failure mode for the emergency condenser is considered acceptable.

2. The present plant Technical Specifications for the emergency condenser permit one tube bundle to be inoperable until the next plant outage if a tube leak develops during plant operation. The last tube leak was experienced in 1973. Since then, a tube bundle has been periodically valved out of service because of outlet valve through leakage. If the other tube bundle was unavailable, the valved out bundle could be used by opening the manual handwheels.

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 systems 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 in this report only with regard to the isolation requirements of the SCS system from the RCS. As discussed in Section 4.2 and 4.3 adequate overpressure protection exists.

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 Big Rock Point valve interlocks, as discussed in Section 5.2, is that adequate interlocks exist.

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 plant to bring the reactor coolant system from a high pressure condition to a low pressure cooling condition.

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 realted systems, to various Design Basis Events, i.e., accidents and transients.

Objective 1(b) relates to availability in the control room of the control and instrumentation systems needed to initiated 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 Big Rock Point Plant to fulfill objective 1(b) is discussed in the preceding sections of this report. Based on these discussions, we conclude that safety objectives 1(b) is met by the safe shutdown systems 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 fire protection reviews are addressing shutdown following a fire in the control room. A local shutdown panel will be installed containing vital instrumentation for use during plant shutdown and cooldown. Suitable procedures for reaching both hot and cold shutdown conditions using the fire protection modifications will be prepared in accordance with 10 CFR Appendix R, item III-L.

The adequacy of the safety grade classification of safe shutdown systems at Big Rock Point, 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 4.1 of this report will be used as input to Topic III-1.

APPENDIX A
SAFE SHUTDOWN WATER REQUIREMENTS

Introduction

Standard Review Plan (SRP) 5.4.7, "Residual Heat Removal (RHR) System" and Branch Technical Position (BTP) RSB 5-1, Rev. 1, "Design Requirements of the Residual Heat Removal System" and Regulatory Guide 1.139 "Guidance for Residual Heat Removal" are the current criteria used in the Systematic Evaluation Program (SEP) evaluation of systems required for safe shutdown. BTP RSB 5-1, Section A.4, states that the safe shutdown system 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. BTP RSB 5-1 Section G, which applies specifically to the amount of auxiliary feed system (AFS) water of a pressurized water reactor available for steam generator feeding, requires the seismic Category I water supply for the AFS to 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. A reasonable period of time to achieve cold shutdown conditions, as stated in SRP 5.4.7, Section III.5, is 36 hours. For a reactor plant cooldown, the transfer of heat from the plant to the environs is accomplished by using water as the heat transfer medium. Two modes of heat removal are available. The first mode involves the use of reactor plant heat to boil water with the resulting steam vented to the atmosphere. The water for this process is typically demineralized, "pure" water stored onsite and, therefore, is available only in limited quantities. The systems designed to use this type of heat removal process (boiloff) are the steam generators for a pressurized water reactor (PWR) or the emergency (isolation) condenser for a boiling water reactor (BWR). The second heat removal mode involves the use of power operated relief valves to remove heat in the form of steam energy directly from the reactor coolant system. Since it is not acceptable to vent the reactor coolant system directly to the atmosphere following certain accidents, the steam is typically vented to the containment building from where it is removed by containment heat removal systems. The containment heat removal systems are in turn cooled by a cooling water system which transfers the heat to an ultimate heat sink - usually a river, lake, or ocean. When using the blowdown mode, reactor coolant system makeup water must be continuously supplied to keep the reactor core covered with coolant as blowdown reduces the coolant inventory. Systems employing the blowdown heat removal mode have been designed into or backfitted onto most BWR's. The efficacy of the blowdown mode for PWR's has received increased staff attention since the Three Mile Island Unit 2 accident in March 1979. Additional studies of the viability of this mode for PWR's are in progress or planned.

This evaluation of cooling water requirements for safe shutdown (and cooldown) is based on the use of the system identified in the SEP Review of Safe Shutdown Systems which has been completed for each SEP facility. The Review of Safe Shutdown Systems used SRP 5.4.7 and BTP RSB 5-1 as a review basis. It should be noted that the SEP Design Basis Events (DBE) reviews, which are currently in progress, may require the use of systems other than those which are evaluated in this report for reactor plant shutdown and cooldown. In those cases, the water requirements for safe shutdown will have to be evaluated using the assumptions of the DBE review.

Discussion

The requirement that a plant achieve cold shutdown conditions within approximately 36 hours, as proffered in BTP RSB 5-1 and SRP 5.4.7, is based mainly on the fact that the amount of onsite-stored water for the AFS of a PWR is limited, and it is desirable to be able to place the RHR system in operation and transfer the plant heat to an ultimate heat sink prior to the exhaustion of the onsite-stored pure water supply. Remaining in a hot shutdown condition, with reactor coolant system temperature and pressure in excess of RHR initiation limits, requires the continued expenditure of pure water via the boiloff mode to remove reactor core decay heat. A BWR relying on the emergency condenser system for cooldown would also be susceptible to the potential exhaustion of onsite-stored pure water.

Should the onsite-stored water supply at a plant be expended, the capability usually exists to use raw water from a river, lake, or ocean, for example, to supply the boiloff systems. However, use of raw water can lead to the degradation, through corrosion, of the boiloff system materials, i.e., steam generator and emergency condenser tubes. This degradation can occur rapidly even if fresh water makeup is used. A plant cooldown and depressurization would help reduce the rate of tube cracking by reducing the stresses in the tube materials. Also, the leakage rate of reactor coolant through potential cracks in the tubes would be reduced if the plant were in a cool, depressurized state.

The original design criteria for the SEP facilities did not require the ability to achieve cold shutdown conditions. For these plants, and for the majority of operating plants, safe shutdown was defined as hot shutdown. Therefore, the design of the systems used to achieve cold shutdown condition was determined by the reactor plant vendor and was not based on any safety concern. Our safe shutdown reviews have pointed out a difference in the vendor approach to system design for cold shutdown. This difference is reflected in the Standard Technical Specification definition of cold shutdown. For a BWR, cold shutdown requires reactor coolant temperature to be $\leq$ 212 degrees Farenheit. For a PWR, cold shutdown requires reactor coolant temperature to be $\leq$ 200 degrees Farenheit. These differences in cold shutdown temperatures require the use of additional systems to achieve cold shutdown for a PWR over and above the systems needed for a BWR. For example, a BWR could use an isolation condenser alone to reach 212 degrees Farenheit (although the approach to 212 degrees Farenheit would be asymptotic); but a PWR, in addition to the steam generators, must use an RHR and supporting systems to get below 200 degrees Farenheit.

Evaluation

After the reactor trip, the reactor system pressure and temperature increase towards the safety valve pressure setpoint because the main condenser is not operable following an assumed loss of offsite power. The emergency condenser is automatically initiated at a reactor pressure of 100 psi above normal, approximately at 1450 psig. The operator is assumed to maintain reactor system pressure near normal operating pressures, by cycling the emergency condenser condensate valves, for a period of four hours prior to commencing the cooldown. The EC capacity is such that a cooldown to SCS initiation conditions can be performed in a reasonable time. The approach to cold shutdown with the EC would be asymtotic.

Emergency condenser pure water makeup is normally supplied by the demineralized water transfer system; the level of makeup water in the emergency condensers is controlled automatically by means of level switches and an air-operated makeup valve. Since the plant compressed air systems are not on the safe shutdown system list, control of the emergency condenser level by manual operation, inside containment, of the makeup supply valve from the fire protection water system may be necessary. As noted before, a plant modification will be implemented to permit control room operation of this valve.

As the cooldown progresses, the reactor system fluid contracts and the need for reactor system makeup exists to keep the level of coolant in the steam drum. If the emergency condenser is used to accomplish the depressurization, the shrink will not uncover the core even if no makeup is provided. The reactor feed system, which is normally used to inject water into the reactor at high pressure is not available because it depends on offsite power. The Control Rod Drive hydraulic system, which can also supply high pressure water, is not considered to be available because it was not designed as a safety system and, therefore, is not included on the safe shutdown system list. Without these high pressure reactor makeup systems, the operator would rely on the core spray (CS) system to supply reactor coolant, if needed. The CS system is a low pressure system (100 psig); and, therefore, if reactor pressure is not below 100 psig, the operator must initiate or permit automatic initiation of the Reactor Depressurization System (RDS) to lower the pressure sufficiently for CS flow into the reactor system to occur. In fact, the RDS can be manually initiated at any time during the cooldown sequence following reactor trip, provided the reactor vessel level at RDS initiation is at or above the RDS automatic actuation level; and the CS system will provide adequate core cooling.* Thus, the RDS and emergency condenser are redundant to each other for the function of plant cooldown. The main reasons that the emergency condenser is included on the safe shutdown list are to provide a core cooling method which does not reduce the reactor system coolant inventory since Big Rock Point does not have the high pressure coolant injection capability that most other boiling water reactors have and because use of the RDS would require extensive cleanup of the containment building.

Normally, long term heat removal would be accomplished by the Shutdown Cooling System (SCS). If this system and its auxiliary systems are available, it would be started at a reactor system pressure of ~200 psig. However, since the SCS initiation requires operator action inside containment and its auxiliaries were not designed and constructed with the quality of the plant safety systems, the RDS, CS, and containment cooling systems (Post Incident Cooling System) would be relied on for long-term cooling of the plant. The core heat and stored heat in the reactor system materials is transferred to the containment by the CS and RDS. The containment heat removal systems transfer the heat to the ultimate heat sink.

Based on the staff's evaluation of safe shutdown water requirements at Big Rock Point, we have concluded that 1) the fire protection water system provides a virtually unlimited supply of makeup water for the emergency condenser, and 2) because of the RDS, CS, and Post-Incident Cooling System capabilities, the plant systems permit a cooldown to cold shutdown conditions in accordance with BTP RSB5-1 requirements.

*Big Rock Point or Reactor Depressurization System Description, Operation, and Performance Analysis, August 15, 1974.