Repro 10/18

October 7, 1982

Docket No. 50-409
LS05-82-10-021

Mr. Frank Linder
Genenal Manager
Dairyland Power Cooperative
2615 East Avenue South
LaCrosse, Wisconsin 54601

Dear Mr. Linder:

SUBJECT: 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) LACROSSE BOILING WATER REACTOR (LACBWR)

A draft evaluation of safe shutdown systems for LACBWR was transmitted to you by letter dated December 22, 1980. Your letter of August 26, 1982, provided comments on this evaluation.

Enclosed is the final topic evaluation for the systems aspects of safe shutdown requirements. The electrical, instrumentation and control aspects of this review, and resolution of your comments in this area will be addressed in a separate evaluation.

As noted in the evaluation, the following will be addressed in the Integrated Assessment:

1) Safety-grade classification of safe shutdown systems (in conjunction with Topics III-1 and III-6).

2) The need for improved shutdown condenser shell side level control.

3) The need for procedures to address provision of demineralized water from onsite Unit 3, contingencies for failure of the shutdown condenser level controller, and use of the manual depressurization system and alternate core spray as a backup method.

SE04
DSU USE Ex (38)
Add: Gary Staley

OFFICE ▶							
SURNAME ▶							
DATE ▶							

This evaluation will be a basic input to the Integrated Safety Assessment for your facility unless you identify changes needed to reflect the as-built conditions of your facility. This topic evaluation 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.

Sincerely,

Original signed by:

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

Enclosure:
As stated

cc w/enclosure:
See next page

OFFICE	SEPB	SEPB	SEPB	SEPB	ORB#5	ORB#5	AD:SA:DL
SURNAME	EMcKenna:bl	TMichaels	CGrimes	WRussell	RDudley	DCrutchfield	FMiraglia
DATE	9/17/82	9/22/82	9/27/82	9/27/82	10/1/82	10/2/82	10/7/82

Mr. Frank Linder

cc
Fritz Schubert, Esquire
Staff Attorney
Dairyland Power Cooperative
2615 East Avenue South
La Crosse, Wisconsin 54601

O. S. Heistand, Jr., Esquire
Morgan, Lewis & Bockius
1800 M Street, N. W.
Washington, D. C. 20036

Mr. John Parkyn
La Crosse Boiling Water Reactor
Dairyland Power Cooperative
P. O. Box 275
Genoa, Wisconsin 54632

Mr. George R. Nygaard
Coulee Region Energy Coalition
2307 East Avenue
La Crosse, Wisconsin 54601

Dr. Lawrence R. Quarles
Kendal at Longwood, Apt. 51
Kenneth Square, Pennsylvania 19348

U. S. Nuclear Regulatory Commission
Resident Inspectors Office
Rural Route #1, Box 276
Genoa, Wisconsin 54632

Town Chairman
Town of Genoa
Route 1
Genoa, Wisconsin 54632

Chairman, Public Service Commission
of Wisconsin
Hill Farms State Office Building
Madison, Wisconsin 53702

U. S. Environmental Protection
Agency
Federal Activities Branch
Region V Office
ATTN: Regional Radiation Representative
230 South Dearborn Street
Chicago, Illinois 60604

James G. Keppler, Regional Administrator
Nuclear Regulatory Commission, Region III
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Mr. Ralph S. Decker
Route 4, Box 190D
Cambridge, Maryland 21613

Charles Bechhoefer, Esq., Chairman
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. George C. Anderson
Department of Oceanography
University of Washington
Seattle, Washington 98195

TABLE OF CONTENTS

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 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)
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 May 22-24, 1978, afforded the team the opportunity to obtain current information and to examine the applicable equipment and procedures, and it also gave the licensee, Dairyland Power Cooperative, the opportunity to provide input into the review.

The review included specific system and equipment requirements for remaining in a hot shutdown condition (> 212°F) and for proceeding to a cold shutdown (≤ 212°F). The review for transition from operating to hot standby 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 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 VII-3 (Systems Required for Safe Shutdown) was completed except for determination of design adequacy of the systems.

Topic IX-3 (Station Service and Cooling Water Systems) was only reviewed to consider redundancy and to a limited extent 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 in support of this topic will be used to resolve the 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"; 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 and are used in the review of facilities being processed 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 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-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 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 electrcial power distribution, both of which will be reviewed later.

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 the event and that of determining the consequences of events. As examples, the safe shutdown topics below pertain to the listed DBEs (the extent of applicability will be determined during the DBE review for La Crosse BWR).

Completion of the safe shutdown topic review (limited in scope as noted above), as documented in this report, provides significant input in assessing the existing safety margins.

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 Coolant Flow (Primary Pump Rotor Seizure) (Primary Pump Shaft Break)	Probability
	VII (Defined Above)	Consequences
X	II (Loss of External Load) (Turbine Trip) (Loss of Condenser Vacuum) (Steam Pressure Regulator [Closed]) (Loss of Feedwater Flow) (Feedwater System Pipe Break)	Consequences
	III (Defined Above)	Consequences
	IV (Defined Above)	Consequences
	V (Defined Above)	Consequences
	VII (Defined Above	Consequences

*For a listing of DBE groups and generic topics, see Reference 5.

2.0 DISCUSSION

2.1 Normal Plant Shutdown and Cooldown (Offsite Power Available, All Equipment Operable)

The plant conducts a controlled shutdown in accordance with written procedures by reducing generator load gradually with the main steam bypass valve closed and turbine inlet pressure controlled at 1225 psig. The initial pressure regulator is in "automatic". *Reactor power* is decreased by control rod insertion at *less than or equal to 15% power per hour*. The regulator closes down the inlet valve to maintain 1225 psig and feedwater flow is controlled in automatic. At about 10% power, station load is transferred to the reserve auxiliary transformer and control rods are inserted to achieve about 2 MWe power at which time the turbine is tripped. *Normally, all control rods are then fully inserted, though the option exists to remain in the low heating range if hot standby is desired.* Feedwater is automatically reduced to match steam flow and secured when this flow reaches its minimum value. These operations minimize perturbations and require minimum actions by operators.

Seal injection adds on the order of *10-15* gpm of subcooled water to the reactor coolant inventory. *The purification system cools, purifies and recirculates approximately 40 gpm.* Steaming continues to the steam jet air ejectors and gland seal thus removing heat and water. The decay heat blowdown valve is manually controlled to blow down via the decay heat removal system to the main condenser to maintain constant inventory.

When proceeding to cold shutdown, the air ejectors are secured; main condenser vacuum is reduced to atmospheric pressure; and the main steam isolation *valve is* closed to minimize the cooldown rate. At a reactor coolant system (RCS) temperature of 470°F, the Decay Heat Cooling System may be placed in service circulating reactor coolant through the tube side of the decay heat cooler. Component cooling water is circulated through the shell side and is in turn cooled in the shell side of the component cooling water coolers by circulating low pressure service water through the tube side. The cooling path to the Mississippi River is thus established and maintained to complete cooldown and remove decay heat while cold. Cool down rates are controlled to maintain reactor vessel temperatures in a range to avoid excessive stresses.

2.2 Shutdown and Cooldown with Loss of Offsite Power

On loss of offsite power the main condenser is unavailable for heat removal for cooldown and the feedwater pumps cannot be used for reactor coolant makeup. After the reactor is tripped, the shutdown condenser is activated manually, automatically by closure of either MSIV, or automatically by system pressure 1325 psig or greater. A cooling path is established by *the* opening *of* two *redundant* inlet valves and two *redundant* condensate return valves.

The shutdown condenser is a closed loop which establishes natural circulation by condensing steam boiled off from the reactor vessel in the tube side of the condenser and returning the condensed steam via gravity flow to a feedwater line and then to a reactor forced circulating loop. Being a closed loop the need for makeup water is minimized. Makeup water is not necessary to maintain adequate core cover. The condenser can transfer *at least 10% of rated reactor* power from the reactor coolant to the shell side water which is boiled off to the atmosphere. The condenser shell side water level is controlled to provide makeup from the demineralized water storage tank and at a lower level (if demineralized system supply is insufficient, exhausted or fails) by high pressure service water. The demineralizer water transfer pumps are powered from the diesel backed essential buses and are capable of adding up to 30,000 gallons of water stored in the virgin water storage tank to the condenser as needed.

Additional demineralized water may be obtained from the adjacent Unit 3 by making a flexible hose connection. Normally, however, the lower controlled level is reached, and high pressure service water system automatically provides makeup using either of two diesel powered pumps which take suction directly from the river. *The Emergency Service Water Supply System could be deployed to provide high pressure service water if neither diesel powered pump was available.* Redundant and diverse methods of makeup water with onsite or offsite power are thus available to the shell side of the condenser. The reactor coolant system cooldown rate may be controlled by controlling the position of the steam inlet valves. The system will cool the RCS *almost* to cold shutdown (about 212°F) and maintain it there indefinitely. Needed instrumentation is powered from the essential busses and provided in the control room.

The High Pressure Core Spray System (HPCS) can also be used for cooling the core. The HPCS can be activated manually or automatically on a low water level signal (-12 inches). Operation of one of two HPCS pumps is required. The water supply is the Overhead Storage Tank (OHST), which has a capacity of *greater than 42,000 gallons. A minimum of 15,000 gallons must be maintained when HPCS is required to be operable. A low level on the OHST will cause automatic transfer of water from the virgin water storage tank, via one of the two demineralized water transfer pumps. Each HPCS pump is capable of supplying 50 gpm against a reactor pressure of 1450 psig to a core spray bundle which discharges the water to nozzles directly above each individual fuel assembly.*

Another method of cooling is available, but is to be used only if other methods fail. This method requires the activation of the manual depressurization system (MDS) and subsequent use of alternate core spray. This is not included as a means for removal of decay heat in plant procedures since venting to containment requires plant downtime for cleanup and restoration to normal conditions. Use of alternate core spray has its primary purpose responding to loss of coolant events. The depressurization system contains two vent valves which open to the containment atmosphere. This function can be performed only if part of the shutdown condenser system is activated, i.e., the venting is performed by passing steam through the shutdown condenser steam inlet valves to and through *one of the two parallel* reactor *emergency* vent valves. Each vent valve is operated from the control room by its independent manual switch. The vent valves open on loss of *nitrogen and fail closed on* loss of power to the solenoids.

Following *use of MDS and* the decrease of system pressure to less than 150 psig, the alternate core spray is activated and delivers ample flow (about 900 gpm at a vessel pressure of 50 psig). This water is provided by two separate diesel driven pumps that take suction from the river and discharge through valves to a 4" line which penetrates the reactor vessel head. This water flows through the open end 4" pipe on to the high pressure core spray system tube bundle, through the flow deflector plate area and downward to the core. Note that operation of this system is procedurally required if the vessel water level is low enough *during a LOCA* coincident with containment pressure of 5 psig. The manual depressurization and alternate core spray systems are designed to comply with the interim acceptance criteria for ECCS. (Reference No. 1)

In addition, the Emergency Service Water Supply System can be used to provide cooling water to the Alternate Core Spray System if the diesel driven pumps fail. Three portable pumps can be connected to either an inside or outside manifold to provide river water to the ACS line inside the Turbine Building.

3.0 CONFORMANCE WITH BRANCH TECHNICAL POSITION 5-1 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.

3.1 Background

A "safety grade" system is defined, in the NUREG-0138 (Reference 2) 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 for Nuclear Power Plant Protection Systems (IEEE 279). The La Crosse Plant was built and received its Provisional Operating Authorization 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). Also, proposed IEEE-279, dated August 30, 1968, was not used in the design of the facility. Therefore, for this evaluation, the systems that should be "safety grade" systems are the systems identified in Table 3.1 and in Section 3.2.

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 safe shutdown systems with RG 1.26. The classification of all systems important to safety for the La Crosse BWR will be determined under SEP Topic III-1, "Classification of Structures, Systems, and Components (Seismic and Quality)." Table 3.1 of this report will be used as input to Topic III-1.

At the time the La Crosse BWR Plant was first licensed, the NRC (then AEC) criteria for QA were not developed. However, the QA program for operation of La Crosse has been 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.

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 on the La Crosse Plant 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 at the La Crosse 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.

The La Crosse fire protection reevaluation resulting from the Browns Ferry fire is currently being reviewed under Multi-plant Action B-41. The results of this reevaluation will be integrated into the SEP assessment of the La Crosse 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."

GDC 5 is not applicable for the La Crosse Station because it does not share any equipment with other facilities.

3.2 Shutdown Systems

The safe shutdown systems which should be "safety grade" are:

1. Reactor Protection and Control Systems (no discussion included)
2. Shutdown Condenser
3. *High Pressure Core Spray*
4. Manual Depressurization System (MDS)
5. Alternate Core Spray (ACS)/*High Pressure Service Water**
6. Reactor Building Main Steam Line Isolation Valve
7. Instrumentation for the Above Systems and Equipment (See Table 3.2)
8. Emergency Power (AC and DC) for the Above Systems and Equipment

---

* A single check valve isolates the Low Pressure Service Water System (LPSW) from the ACS *and High Pressure Service Water Systems* (HPSW). Therefore, this system should be of the same seismic and quality classification as the ACS and HPSW. *The Low Pressure Service Water System and portions of the HPSW System* are not required to function for safe shutdown.

In addition to these systems, other systems may function as backup for the above systems and components. Some of the backup components are discussed in this and other sections of the report to identify alternate ways that may provide an acceptable level of safety.

Shutdown Condenser System

The shutdown condenser system provides the capability to take the reactor from hot shutdown to cold shutdown; i.e., BTP 5-1 Functional Requirement No. 1, is described below:

The shutdown system consists of the shutdown condenser, piping, valves, and associated instrumentation.

The shutdown condenser is located on a platform 10 feet above the main floor in the reactor building. Steam from the 10-inch main steam line passes through a 6-inch line, two parallel inlet steam control valves, back to a 6-inch line and into the tube side of the condenser where it is condensed by evaporating cooling water on the shell side. The steam generated in the shell is exhausted to the atmosphere through a 14-inch line which penetrates the reactor building. An area monitor is located next to the steam vent line near the containment shell penetration in order to detect excessive activity release in the event of a shutdown condenser tube failure. The main steam condensate is collected and returned to the reactor vessel by gravity flow. The condensate line leaving the condenser is a 6-inch line along the horizontal run and is reduced to a 4-inch line for the remainder of the vertical section. Two parallel condensate outlet control valves are located in the 4-inch return line. The condensate line also contains two 2-inch vent lines which join together and return to the lower channel section of the condenser. The vents are provided for returning vapors and/or noncondensible gases which are carried into the condensate line back to the condenser to prevent perturbations in the condensate flow leaving the condenser. The lower channel section in turn is vented to the off-gas system through a 1-inch vent line. Flow in this vent line is restricted by a 1/16-inch orifice, which is built into and is an integral part of the shutdown condenser off-gas control valve seat.

A vent line containing two parallel control valves is connected to the 6-inch condensate return line. The valves discharge directly to the reactor building atmosphere and can be used, in an emergency, to vent air *or steam* from the reactor vessel in the event that it should become necessary to flood the reactor building due to a large leak below the reactor core. This would permit the water level *and pressure* in the building to equalize with that in the reactor vessel.

The shell, steam inlet channel, condensate outlet channel, and tube sheet are made of carbon steel. All parts in contact with the reactor steam are clad with monel, except the tubes which are made of 70 percent copper and 30 percent nickel. The shutdown condenser is designed to absorb, without damage, the thermal and physical shock of going from ambient temperature to full load conditions in 5 seconds for 50 cycles during a 20-year unit lifetime. The thermal shock is equivalent to a temperature transient of 500°F in 5 seconds.

The tubes are seal welded to the tube sheet and the tube sheet is welded to the shell; however, a cutting ring is provided for tube bundle removal should the need arise.

Due to the large temperature differential between the reactor steam and the shell side cooling water, a thermal barrier is provided to reduce thermal stresses in the tube sheet. The barrier consists of a shield plate and individual tube ferrules which insulate the steam inlet channel and tube sheet from the cold water. The large temperature differential also has led to the use of finned tubes. The finned tubes present an irregular surface on the outside tube, which limits film boiling by breaking up the film. Film boiling tends to insulate the tubes, thus decreasing the heat transfer coefficient. The tube surface area is 3705 square feet and will handle 39.3×10^6 Btu/hr at a transfer rate of 27.7 Btu/hr/sq.ft/°F.

The system piping is designed for a maximum working pressure of 1400 psig at 650°F. *Primary* system operating pressure is ~ 1250 psig. The steam piping from the biological shield to the inlet of the shutdown condenser is constructed of Schedule 120 low-alloy steel. All other steam and condensate piping are of Type 304 stainless steel. The steam piping within the biological shield and the 6-inch section of the condensate return line is Schedule 120 and the 4-inch section is Schedule 80.

Valves in the system meet and are in accordance with ASME codes and standards.

The condenser has 2 -6" steam inlet angle valves in parallel. They are air operated, air to close, controlled by 125V DC control power. They fail open on loss of power and may be manually vented *(opened)* at the valve station on the platform at the shutdown condenser (about *3* minutes from the control room). The solenoids for control air to each valve are provided from separate DC sources. The position of each valve is indicated in the control room by a hand indicator-controller by selection and by indicating panel lights.

The condenser condensate return to reactor has 2 -4" angle valves in parallel. They are air operated, air to close, controlled by 125V DC control power. They fail open on loss of DC or air and may be manually vented *(opened)* at the valve station on the platform at the shutdown condenser (about 3 minutes from the control room). The solenoids for control air to each valve are provided from separate DC sources. The position of each valve is indicated in the control room by panel lights.

The only instrumentation required to know the status of Shutdown condenser cooling is reactor coolant system pressure. Redundant measurements are provided in the control room. Also, secondary side water level is provided in the control room and at the condenser platform. Other indications in the control room are tube side vent pressure, shell and tube side temperature, valve positions, and controls.

Reactor pressure channels 1 or 2 will automatically activate the condenser system at a pressure of 1325 psig. Closure of either the *Reactor Building Main Steam Isolation Valve or the* Turbine Building Main Steam Isolation Valve will automatically activate the shutdown condenser system. Simultaneous failure of either valve closure circuits (2) will result in high reactor pressure and activation of both channel 1 and 2 reactor pressure protection circuits and the condenser. Thus the protection is redundant and diverse on isolation valve closure.

The *shutdown* condenser system may be manually activated.

Secondary side water level is indicated locally by two gauge glasses; remote indication is provided by an air to current converter. The current to the indicator in the control room also provides high and low level alarms. The air supply is provided by plant air system. The level controller provides a 3 - 15 psi air signal split, 3 - 9 psi to high pressure service water system makeup valve and 9 - 15 psi for the demineralized water makeup valve.

Secondary side makeup water is provided by the demineralized water system with the high pressure service water (HPSW) system providing a backup supply.

The demineralized water transfer system is the normal source of secondary side makeup water to the shutdown condenser. The system consists of the demineralized water storage tank, two transfer pumps, piping, valves, and associated instrumentation.

The demineralized water storage tank, located on the roof of the turbine building, has two compartments. The topic compartment has a capacity for ~30,000 gallons of virgin demineralized water. The bottom compartment contains potentially radioactive condensate from the main condenser hotwell and has a capacity of 19,000 gallons. The water supply to the demineralized water storage tank consists of two deep-well 120 gpm pumps capable of supplying 40 gpm via the demineralization train. The pumps are non-essential loads with no backup power source available.

There are two demineralized water transfer pumps that maintain the demineralized water transfer lines continually pressurized to supply water on demand. The transfer pumps are powered from separate essential buses that are diesel generator backed.

The HPSW system is the backup source of secondary side makeup water to the shutdown condenser. In addition, the HPSW system can supply the high pressure core spray (HPCS) system, the ACS system, the fire protection system, the circulation water fall-out contamination monitor eductor, and the crib house screen wash system.

The HPSW non-essential powered motor-driven pump located in the turbine building takes positive suction at ~65 psig from the LPSW system. The HPSW pump discharges into a header that divides into two main loops. One loop serves the turbine building, the containment building, the waste treatment building, and the gas vault. The other loop supplies the outside fire protection system and the crib house.

During normal operation, the motor-driven pump, with a capacity of 500 gpm at a 150-ft working head, automatically maintains HPSW system pressure between 75 ± 5 psig and 125 ± 5 psig. The pump is protected against low suction pressure by a 35 psig suction pressure trip. HPSW system pressure fluctuations are mitigated by the air space in the 1,400-gallon HPSW storage tank. The HPSW storage tank serves as an accumulator for the system and can be air-loaded to a working pressure of 150 psig. The HPSW pump is backed up by two pumps driven by separate diesel engines, the alternate core spray pumps, *taking suction directly from the river. The Emergency Service Water Supply System pumps can be deployed to provide HPSW if the diesel driven pumps are not available.*

The makeup water system relies on a single controller. Although the makeup water is provided from redundant sources each with its own piping and valves, a single failure of this controller in a fashion to keep air on the control valves would not indicate the failure in the control room, nor would makeup be provided. Following boil off of the contained water, system pressure would begin to rise. I: the operator could not go to the condenser platform inside containment in time, the safety valves would open. *The makeup valves would fail open on loss of air or loss of power.*

Air is supplied to the shutdown condenser valves and the level controller from the containment building control air by the station air system. This system has two two-stage compressors. The system air providec for control comes from these compressors and is filtered and dried. This air is backed by a *backup air compressor that consists of* two compressors driven by a single motor. These compressors come on at a drop from a normal 100 psig to 75 psig. An adequate air supply is thus provided.

With only onsite power available, the backup instrument air system is the only source available since it is the only compressor powered from the essential bus. It is needed for operation of the secondary side level controller; however, on loss of instrument air the makeup valves fail open and provide a continuing makeup supply to the secondary side.

High Pressure Core Spray

The High Pressure Core Spray (HPCS) System can also be used for core cooling during a loss of offsite power. There are two redundant HPCS pumps, supplied from separate essential buses. During plant operation, the system is lined up to provide water to the vessel. To start water flowing to the core, one of the pumps needs to start, no valves need to be repositioned.

The HPCS System is capable of supplying water to the reactor at all primary pressures, up to and including relief valve lifting pressures.

Alternate Core Spray System

Alternate Core Spray (ACS) in conjunction with the manual depressurization system (MDS) is a redundant *method of cooling the core.* Its use in this mode is not included in plant procedures. To use alternate core spray requires reactor pressure to be below 150 psig. Therefore, the vent valves must be opened and pressure reduced if the shutdown condenser is not functioning. Heat added to the primary containment by the vent valves is transferred to the atmosphere through the containment dome wall. The vent valves are manually operated in this situation. They are separate and redundant and no single failure will prevent the manual depressurization system from performing its function (Reference 1). However, to use either vent valve requires *at least one shutdown condenser system steam inlet valve to be open.* These operations are controlled manually, understood by senior operators and are incorporated into the major primary leak plant procedures, *but not the shutdown or loss of power procedure.* The reason they are excluded is because the consequences of unneeded venting are severe and would require significant interruption in operations.

Emergency Service Water Supply System

The Emergency Service Water Supply System (ESWSS) provides a backup method of supplying High Pressure Service Water to the Alternate Core Spray line. The system is manually deployed. Three gasoline engine driven portable pumps are connected in parallel with the discharge hose connected to the High Pressure Service Water line via either an inside or outside manifold.

Decay Heat Cooling System

Following a normal reactor shutdown the decay heat cooling system is used to cool the reactor to 120°F. Usually the main condenser is preferred for decay heat removal until the RCS is considerably cooler than the allowed 470°F decay heat cooler initiation temperature because the cooling rate can be controlled better. After initiation the *Decay Heat System* is used to maintain the reactor water temperature below 120°F, *including* while the reactor vessel is open for refueling or alterations. Also it can be used to provide additional heat to the reactor to satisfy loop piping Nil Ductility Transition (NDT) temperature requirements and provides a path via the blowdown line to remove excess reactor water from the reactor to the main condenser.

The decay heat cooling system takes its suction from the *reactor side of the isolation valve on the* inlet line to the forced circulating pump 1A, reactor water then flows to the decay heat pump which discharges to the tube side of the decay heat cooler and returns it to the reactor side of the forced circulation pump isolation valves. The shell side of the decay heat cooler is cooled by the component cooling water system which is completely redundant in *active* equipment and power supply. The component cooling water system heat exchangers shell sides are cooled by the *low pressure* service water system which obtains its water from the Mississippi River. Since the LPSW system is not powered from the essential buses, a loss of offsite power renders the LPSW pumps inoperable thereby interrrupting the heat exhange from the decay heat cooling system to the river. To reestablish the continuity of heat exchange the component cooling system heat exchanger shell side can be connected via a *cross-connect line, to the High Pressure Service Water System* which utilizes two diesel driven pumps that also receive their water supply from the Mississippi River. This is covered in operating procedures.

The decay heat pump is a single stage centrifugal pump powered from the essential bus diesel backed and fed through reactor building motor control center 1A. It is designed for continuous operation at 1500 psig and 585°F.

Control stations for the pump are located in the control room and locally at the pump, indication of pump status is provided at both locations. In addition to the pump status indications (*red* light-pump energized, *green* light-pump deenergized, white light-auto trip), an audible alarm annunciates in the control room whenever the automatic trip functions.

The decay heat cooler can serve as a cooler (normal mode) or as a heater. In the cooling mode, reactor water enters the tube side and makes two passes then exits, component cooling water enters the shell side also making two passes. In the heater mode, which is used to facilitate loop NDT requirements, heating steam enters the shell side.

The shell side of the cooler is rated by design at 150 psig and 375°F, the tube side is rated by design at 1500 psig and 470°F. The system design limit of 470°F is based on thermal stressing of the cooler tube sheet.

All piping and valves in the decay heat system with the exception of the blowdown line are designed to 1400 psig and 595°F and are made of 304 stainless steel. The blowdown line which is used to maintain a constant volume of water in the reactor is made of carbon steel and is designed to 1450 psig and 650°F.

The blowdown valve which is a 2-inch, 1500 psig, air operated valve can be controlled locally by handwheel or remotely from the control room. Approximately *10* gpm of subcooled water enters the primary system through the seals of the forced circulation pumps and rod drive mechanisms. During operation this *10* gpm serves as makeup water and is of no consequence, however, during shutdown when normal feedwater is reduced, the *10* gpm becomes excessive; therefore, the decay heat system is utilized to blowdown the excess inventory to the main condenser. The blowdown valve is electrically interlocked with the reactor protection system such that a low reactor water level scram will close the blowdown valve to preclude further reduction of reactor water level. The blowdown valve fails closed on a loss of control air or electric power thereby removing any possibility of draining the reactor.

During local operation of the blowdown valve, reactor *water level* is available for the operator to monitor. During remote operation the operator has available all of the parameters in the control room. Local operation is required during shutdown from outside the control room.

The decay heat system blowdown line can also be used during a "Feed and Bleed" operation using the HPCS pumps to feed the system and the blowdown line to reduce and maintain the reactor water level at a predesignated level. Thus the *primary system* can be cooled by this mode as an alternate cooldown method.

Although the decay heat system is normally used during routine shutdown of the reactor, it has no redundant components and lacks redundant power supplies; however, the function of cooldown is provided for by use of the shutdown condenser, *high pressure core spray system* or manual depressurization and alternate core spray systems thereby providing redundant diverse methods for cooldown.

We have concluded that all operations needed for safe shutdown with offsite or only onsite power available can be done from the control room. Nevertheless, with credit for limited action outside the control room, the shutdown condenser reliability can be improved. By dispatching an operator to the platform inside containment, the single failure vulnerability of the secondary side level controller can be overcome. The *double sight glass* shows water level, the makeup valves can be adjusted to provide makeup water and reactor pressure and water level are indicated in the control room. However, it should be noted that the condenser does not have to be immune from single failure considering the redundant and diverse depressurization coupled with alternate core spray *and the high pressure core spray system,* which provide alternate shutdown and cooldown methods without offside power. Likewise, in the temperature range below 470°F, the Decay Heat Cooling System can be used to cooldown and remove decay heat while cold. Manual operations are required for the loss of offsite power situation.

The shutdown condenser will bring the reactor to *almost* cold condition from operating conditions. The condenser relies on boil off of secondary water to accomplish cooldown. Therefore, it will bring the RCS temperature to *approximately* 212°F. The heat removal capacity is so large (equivalent to > 10% of rated power) that steam inlet flow must be controlled to avoid excessive thermal stress to the reactor vessel. Even with the failure of the secondary side level controller, and dispatch of the operator to the platform, the rate of cooldown will have to be controlled to avoid excessive thermal stresses. The controlled cooldown rates are reasonable.

When the manual depressurization and alternate core spray systems are used to cool down, the vent valves provide a rapid depressurization to atmospheric pressure and the alternate core spray will cool the core indefinitely. This sytem has been analyzed by the staff and found to be acceptable for emergency core cooling during a loss-of-coolant accident with only onsite or offsite power available and with the most limiting single failure.

The capacity of the combined cooling of the shutdown condenser ($\geq$ *10%* of rated power) from operating temperature (or alternatively high pressure core spray or manual depressurization with the vent valves) to 470°F and the Decay Heat Cooling System below 470°F with a heat removal capability about 1/6 that of the shutdown condenser, the cooldown can be accomplished. This alternative will require longer to cooldown since *high pressure service* water must be connected to provide cooling to the secondary side of the component cooling heat exchanger.

3.3 Safe Shutdown Instrumentation

Table 3.2 lists the instruments required to conduct a safe shutdown. The list includes those instruments which provide information to the control room operator from which the proper operation of all safe shutdown systems can be inferred. These instruments are the RCS pressure ~~and temperatures,~~ reactor vessel level, and shutdown condenser secondary side water level. Improper trending of these parameters would lead the operator to investigate the potential causes. Other instruments listed in the table provide the operator with (1) a direct check of safe shutdown system performance and (2) an indication of actual or impending degradation of system performance. The design of the instrumentation and controls used for safe shutdown is evaluated in the electrical portion of the resolution of SEP Topic VII-3.

TABLE 3.1 CLASSIFICATION OF SHUTDOWN SYSTEMS - LA CROSSE BOILING WATER REACTOR

System, Subsystem, Component	Quality Group R.G. 1.26	Quality Group Plant Design	Seismic R.G. 1.29	Seismic Plant Design	Remarks
Shutdown Condenser (tube side)	ASME III Class *2*	ASME VIII Case 1270N	Category I	Note 1	Note 1: Although not originally designed to withstand a seismic event, an analysis of the LACBWR concluded that the facility could withstand an earthquake with corresponding maximum horizontal ground acceleration of 0.12g
(shell side)	ASME III Class 3	ASME VIII Case 1270N	Category I	Note 1	
Piping from reactor vessel to shutdown condenser up to and including safety valves, main steam isolation valve 64-25-030, main feed shutoff valve, MDS valves (62-25-013, 014) and vent and drain lines larger than 1" diameter.	ASME III Class 1 & *2*	*ASA B31.1 (1955)*	Category I	Note 1	
Vent and Drain piping smaller than 1" diameter.	ASME III Class 2	*ASA B31.1*	Category I	Note 1	Footnote 2(a) to 10 CFR 50.55a.
Overhead Storage Tank (OHST)	ASME III Class 2	AWWA D-100	Category I	Note 1	
High Pressure Core Spray System	*ASME III Class 1 & 2*	*ASA B31.1*	*Category I*	Note 1	
Alternate Core Spray (ACS) Pumps (2 diesel driven)	ASME III Class 2 & *3*	*S & L Spec. W 1924*	Category I	Note 1	These pumps are same as diesel fire pumps.
Diesel engine fuel supply	ASME III Class *3*	*S & L Spec. W 1924*	Category I	Note 1	Portions of steam line are used for both MDS and Shutdown Condenser

TABLE 3.1 CLASSIFICATION OF SHUTDOWN SYSTEMS - LA CROSSE BOILING WATER REACTOR

System, Subsystem, Component	Quality Group R.G. 1.26	Quality Group Plant Design	Seismic R.G. 1.29	Seismic Plant Design	Remarks
Piping from pumps to outermost containment isolation valve up to and including relief valves and HPSW supply, drain vent, and test line isolation valves, strainers, and valves which isolate non-essential portions of the system.	ASME III Class 2 & 3	*ASA B31.1(55)* *ASA B16.9(58)* *ASA B16.10(57)* *ASA B16.5 (61)*	Category I	Note 1	
Piping from outermost containment isolation valves up to reactor vessel including vent piping greater than 1" diameter.	ASME III Class 1 & 2	*B31.1*	Category I	Note 1	
Process Instrumentation and Controls	NA	--	Category I	Note 1	
Emergency Power Supply System	NA	--	Category I	Note 1	
Diesel generators			Category I	Note 1	
DC power supply systems			Category I	Note 1	
Distribution lines, switchgear, motor control centers			Category I	Note 1	
Reactor Control and Protection Systems	NA	--	Category I	Note 1	

TABLE 3.2 SAFE SHUTDOWN INSTRUMENTS

Component/System	Instrument Indication	Instrument Location	REFERENCES
Control Rod Drive System	Control Rod Position	RPT-Inside containment RPI-Control room	LACBWR Safeguards Report (SR) Sec. 8.5
Reactor Coolant System	RCS Pressure (PT 50-35-301 & PT 64-35-301, 302)	PT-Inside containment PI-Control room	LACBWR SR Fig. 5.1
	Coolant Level in Core (LT 50-42-301, 302, 303)	LT-Inside containment LI-Control room	LACBWR SR Fig. 5.1
Shutdown Condenser System	Shutdown Condenser Shell-Side Water Level (LG 62-42-801, 802, 804)	2 local gage glasses inside containment. Level indication and alarms in control room.	LACBWR SR Fig. 5.6 LACBWR Operating Manual (OM) Vol. II, Sec. 5.3
	Shutdown Condenser System Valve Position Indication	VPI-Control room for major system valves	LACBWR OM Vol. II, Sec. 5.3
High Pressure Service Water System	HPSW Storage Tank Level (Sight Glass 75-98-001)	Local level indication. High/low level alarms in control room	LACBWR OM Vol. V, Sec. 7
	HPSW System Flow to ECS and Shutdown Condenser Systems (PT 75-35-301)	PI-Control room Flow and pressure alarms in control room	LACBWR OM Vol. IV, Sec. 7

TABLE 3.2 (Cont.)

Component/System	Instrument Indication	Instrument Location	References
Alternate Core Spray System	ACS Flow (FT 38-37-301)	Indication in control room	LACBWR OM Vol. II, Sec. 17
High Pressure Core Spray System	HPCS Flow (FS 53-37-70)	Local pressure indication. Low flow alarm in control room	LACBWR OM Vol. II, Sec. 8
Overhead Storage Tank	OHST Level (LIS 69-42-801)	Low level alarm in control room. Local bubbler-type indication	LACBWR OM Vol. II, Sec. 15
Emergency AC Power	480-V Essential Buses 1A and 1B Voltage Indication and Low Voltage Alarm	Breaker trip alarms in control room	LACBWR OM Vol. IV, Sec. 14, Dwg. No. 503621
	120-V Buses Volt Indicating Lights, Alarms, Voltage, and Current	Indication in control room	Dwg. No. ES-58
Diesel Generators	Generator Output (Voltage, Current, Frequency, Power in VARS)	Control room	Dwg. No. 503627
Emergency DC Power	125-V DC Low Voltage Alarm for each feeder from Bus	Control room	Dwg. No. 503627

4.0 SPECIFIC RESIDUAL HEAT REMOVAL AND OTHER REQUIREMENTS OF BRANCH TECHNICAL POSITION 5-1

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 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.

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.

La Crosse Boiling Water Reactor does not have a low pressure redundant residual heat removal system. The system which performs the residual heat removal function for the latter stage of normal cooldown and for long term decay heat removal while depressurized is the Decay Heat Cooling System. Since it is designed for reactor coolant system (RCS) pressure, the isolation requirements listed above do not apply; i.e., the potential for an RCS break from overpressurization because of valving errors is of no safety consequence.

4.2 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 nonisolatable 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 the RCS pressure exceeds the RHR system design pressure, adequate relief capacity shall be provided during the time period while the valves are closing.

The Decay Heat Cooling System does have code required relief valves set at design pressure to protect against overpressurization while isolated, and it is protected from overpressurization while in service by code safeties. Again, it is a full pressure system and protection from accidental overpressurization is adequate with reactor vessel code safety valves. Its heat exchanger is temperature limited to operation below 470°F, and this limit is administratively controlled.

4.3 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 pump is designed for continuous operation at 1500 psig and 585°F. The pump requires a net positive suction head (NPSH) of approximately 10 ft. to preclude cavitation and eventual impeller corrosion. Due to its location it has an available suction head of approximately *50* ft. Cooling water to the pump bearings is supplied by the component cooling water system.

4.4 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.

Based on the previous discussion, the Decay Heat Cooling System does not require isolation and interlock circuits.

4.5 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.

The licensee has procedures to perform safe shutdown operations including shutdown to hot standby, operation at hot standby, hot shutdown, operation at hot shutdown and cold shutdown including long-term decay heat removal. The licensee has also provided the operating staff procedures covering offnormal and emergency conditions for shutting down the reactor and decay heat removal under conditions of loss of system or parts of system functions normally needed for shutdown and cooling the core. Procedures for systems operation for systems used in safely shutting down the reactor are also included in the plant operating procedures. These procedures include provisions identified in Regulatory Guide 1.33. These procedures were reviewed and are in conformance with Regulatory Guide 1.33.

Some operations are not covered that should be addressed are:

1. Providing demineralized water from the onsite Unit 3 to the reactor plant demineralized water system.

2. Contingencies for failure of shutdown condenser secondary side water level controller and loss of air supply to controller.

3. Use of manual depressurization system in conjunction with alternate core spray as a backup safe shutdown system. Alternatively, we believe this method may be covered in the licnesse's retraining program and simply identified as a means of achieving cooldown in procedures, if the licensee desires. He has indicated that the severity of contamination following unneeded use of this method would interrupt plant operation for a long period to clean up the contamination, and he would prefer to omit from procedures to avoid use unless other methods have proven unsatisfactory.

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 La Crosse Plant meets the safety objective of these topics.

5.1 Topic V-10.B RHR System Reliability

The safety objective of 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 La Crosse BWR systems have been compared with the criteria of BTP 5-1 and the results of these comparisons are discussed in Sections 3.0 and 4.0. Section 3.0 discusses the way the functional requirements are met and Section 4.0 discusses the Decay Heat Removal System which performs the function identified in BTP RSB 5-1 as Residual Heat Removal. The Decay Heat Removal System has very limited use in the La Crosse plant and it does not contain system redundancy. However, we have concluded that the other La Crosse systems acceptably fulfill the safety objective subject to the resolution of the following in the SEP integrated assessment:

1. The requirement for using only safety grade equipment to accomplish the shutdown and cooldown capability. The quality and seismic classification of safe shutdown systems identified in Section 3.0 will be established in further review of Topic III-1 and III-6.

2. The need for improved shutdown condenser shell side level control to preclude a single failure disabling the condenser. Resolution of this item can be postponed until the integrated assessment because of the existence of a redundant cooldown method (MDS and ACS).

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

The safety objective of this topic is to assure that adequate measures are taken to protect low pressure systems connected to the primary system from being subjected to excessive pressure which could cause failures with the potential to cause a LOCA outside of containment. The topic in this review is concerned only with the decay heat cooling system; high/low pressure interfaces with other systems were not reviewed. Since this system is completely contained within containment, except for a portion of the blowdown line, and since it is designed for system pressure, the overpressure potential is minimal (i.e., the same as the rest of the RCS); and the topic is resolved for the decay heat cooling system.

5.3 Topic V-11.B RHR Interlock Requirements

The safety objective of this topic is identical with V-11.A. The interlock would close low pressure isolation valves when open and high pressure excursion occurs and would prohibit opening when high pressure exists. Again, this system is designed for full system pressure and the interlocks are unnecessary. This topic is resolved for the decay heat removal system.

In addition to these requirements, and as a matter to be resolved separately from the SEP, the NRC staff has determined that certain isolation valve configurations in systems connecting the high-pressure Primary Coolant System (PCS) to lower-pressure systems extending outside containment are potentially significant contributors to an intersystem loss-of-coolant accident (LOCA). Such configurations have been found to represent a significant factor in the risk computed for core melt accidents (WASH-1400, Event V). The sequence of events leading to the core melt is initiated by the failure of two in-series valves to function as a pressure isolation barrier between the high-pressure PCS and a lower-pressure system extending beyond containment. This causes an overpressurization and rupture of the low-pressure system, which results in a LOCA that bypasses containment.

The NRC has determined that the probability of failure of these valves as a pressure isolation barrier can be significantly reduced if the pressure at each valve is continuously monitored or if each valve is periodically inspected by leakage testing, ultrasonic examination, or radiographic inspection. NRC has established a program to provide increased assurance that such multiple isolation barriers are in place in all operating Light Water Reactor plants. This program has been designated DOR Generic Implementation Activity B-45.

In a generic letter of February 23, 1980, the NRC requested all licensees to identify susceptible valve configurations which may exist in any of their plant systems communicating with the PCS. For plants in which valve configurations of concern were found to exist, licensees were further requested to indicate: (1) whether, to ensure integrity, continuous surveillance or periodic testing was currently being conducted, (2) whether any valves of concern were known to lack integrity, and (3) whether plant procedures should be revised or plant modifications be made to increase reliability.

LACBWR is one of those plants identified as being susceptible to the potential failure because of the configuration of valves (two check valves in series) in the Alternate Core Spray System. Therefore, as noted in Reference 6, action has been taken independently of the SEP effort and resolved the "Event V" problem.

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.

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

Objective 1(b) relates to availability in the control room of 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 La Crosse plant to follow the shutdown is discussed in the preceding sections of this report. Based on these discussions and knowledge gained from the site visit, we conclude that safety objective 1(b) is met by the safe shutdown systems, actuation mechanisms, and control room displays, at La Crosse subject to the findings of related SEP Electrical, Instrumentation and Control topic reviews (See References 7 and 8).

---

* BTP 5-1 applies to both PWRs and BWRs.

Safety objective 2 requires the capability to shut down to both hot shutdown and cold shutdown conditions using systems, instrumentation and controls located outside the control room. La Crosse plant has a procedure, "Emergency Reactor Shutdown and Cooldown When the Control Room is Inaccessible." The procedure assumes lack of time to trip reactor prior to leaving the control room and indicates the location from which the reactor can be tripped. It covers emergency communications and locations of portable radios. It designates the operators stations and actions to be taken by them, provides adequate instructions for ascertaining the operability and condition of the essential plant equipment and indicates the surveillance instrumentation and instructions for interpreting the information. The plant relies on the Shutdown Condenser and Decay Heat System to control shutdown and cooldown.

The review team visited each designated operators station and assessed the capability of the plant staff to perform the necessary operations. We conclude that the plant can perform these shutdown operations and indeed has done so. Early in plant life power cables were cut inadvertently and the plant lost all AC power. Operators were dispatched to their local stations and the plant was successfully shut down and controlled.

The adequacy of the safety grade classification of safe shutdown systems at La Crosse, 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 Design Basis Review Event reviews.

5.5 Topic X Auxiliary Feed System (AFS)

The safety objective of this topic is to assure that 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.

This topic is not applicable to La Crosse. However, the cooling water inventory requirements for a safe shutdown of the facility using the systems identified in Section 3 are evaluated in Appendix A.

6.0 REFERENCES

1. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 6 to Provisional Operating License No. DPR-45, Dairyland Power Cooperative, La Crosse Boiling Water Reactor, Docket No. 50-409, August 12, 1976.

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

3. NRC Letter, R. Reid to J. Madgett, dated February 14, 1977.

4. La Crosse Boiling Water Reactor Operating Manual, Volume I through V.

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

6. Amendment 24 to POL No. DPR-45, February 25, 1981.

7. Letter LS05-82-03-059, D. Crutchfield (NRC) to F. Linder (DPC), dated March 9, 1982.

8. Letter LAC-8535, F. Linder (DPC) to D. Crutchfield (NRC), dated August 26, 1982.

APPENDIX A

SAFE SHUTDOWN WATER REQUIREMENTS

Introduction

Standard Review Plan (SRP) 5.4.7, "Residual Heat Removal (RHR) System"; Regulatory Guide 1.139, "Guidance for Residual Heat Removal"; and Branch Technical Position (BTP) RSB 5-1, Rev. 1, "Design Requirements of the Residual Heat Removal System," 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 (PWR) 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 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. 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, water is the medium for transfer of heat from the plant to the environs. Two modes of heat removal are available. The first mode involves the use of reactor plant heat to boil water and the venting of the resulting steam to the atmosphere. The water for this process is typically demineralized "pure" water stored onsite and, therefore, is limited in quantity. The systems designed to use this mode of heat removal (boiloff) are the steam generators for a PWR or the emergency (isolation) condenser for a boiling water reactor (BWR). The second heat removal mode (blowdown) 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, the steam is typically vented to the containment building from which containment cooling water systems transfer the heat to an ultimate heat sink, usually a river, lake, or ocean. When the blowdown mode is used, reactor coolant system makeup water must be continously supplied to keep the reactor core covered with coolant to compensate for loss of coolant inventory. The efficacy of the blowdown mode for PWRs has received increased staff attention since the Three Mile Island Unit 2 accident in March 1979. Additional studies are in progress or planned.

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

Discussion

The requirement in BTP RSB 5-1 and SRP 5.4.7 that a plant achieve a cold shutdown condition within approximately 36 hours is based mainly on the desire to activate the RHR system and transfer the plant heat to an ultimate heat sink prior to the exhaustion of the limited amount of onsite-stored pure water available for the AFS of a PWR. A sustained hot shutdown condition, with reactor coolant systems temperature and pressure in excess of RHR initiation limits, requires continued boiling off of pure water 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.

If onsite-stored water at a plant is depleted, raw water from a river, lake, or ocean can usually be tapped to supply the boiloff systems. However raw water can accelerate the corrosion of boiloff system materials in the steam generator and emergency condenser tubes even if the water is fresh. Seawater can cause chloride stress-corrosion cracking of the tubes within 1 week. Raw fresh water can cause caustic stress-corrosion cracking of both stainless steel and inconel tubes in less than 72 hours through NaOH concentration. Plant cooldown and depressurization would help reduce the rate of tube cracking by reducing the stresses in the tube materials and would reduce the rate of reactor coolant leakage through cracks that do occur.

The orginal 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 a cold shutdown condition was determined by the reactor plant vendor and was not necessarily based on safety concerns. Safe shutdown reviews have pointed out a difference in vendor approaches to system design for cold shutdown relected in the Standard Technical Specification definition of cold shutdown. For a BWR, cold shutdown requires reactor coolant temperature to be $\leq$ 212°F; for a PWR, the temperature is $\leq$ 200°F. This difference in cold shutdown temperature requires additional systems for PWR cooling not needed for a BWR. For example, a BWR could use an isolation condenser alone to reach 212°F (although the approach to the final temperature would be asymptotic), but a PWR must use a RHR system and supporting systems, in addition to the steam generators, to cool to 200°F.

Evaluation

Table B.2-1 provides plant-specific data and assumptions used in the staff calculation of safe shutdown water requirements for the LaCrosse Unit 1 plant. Table B.2-2 shows the results of the calculation.

Upon loss of electrical load or turbine trip, the main steam bypass valve, with a capacity of 100% steam flow, will open and dump steam directly to the main condenser. Following a loss of offsite power and turbine trip, the main condenser circulating water pumps cannot be powered from onsite sources. With the loss of the circulating water pumps, the main condenser vacuum cannot be maintained, and the main condenser becomes unavailable for heat removal. With the loss of the normal heat sink and closure of the reactor building steam isolation valve following the reactor scram, the shutdown condenser will actuate to dissipate the decay heat.

The capacity of the shutdown condenser (> 10% of rated reactor power) is sufficient to remove all decay heat generated from the time of the reactor scram. To prevent excessive RCS cooldown rates or to maintain the plant in a hot shutdown condition, the operator must limit the amount of steam going to the shutdown condenser by throttling the shutdown condenser steam inlet valves. Based upon scoping calculations, the reactor is capable of being cooled to 235°F in 36 hours following a 4-hour wait after the reactor shutdown. This calculation used only the shutdown condenser for cooling and assumed a single active failure of the single train decay heat cooling system.

Four hours after the reactor shutdown, the decay heat rate is 5.01×10^6 Btu/h and the integrated decay heat over the 4-hour period is 2.92×10^7 Btu. Assuming that the plant operator has maintained a normal hot shutdown temperature of 577.5°F and a constant mass of water in the reactor vessel during the 4-hour wait period, 3320 gallons of demineralized water have been used to maintain the shutdown condenser shell side water level and remove the integrated decay heat.

Scoping calculations indicate that if the maximum inlet steam flow to the shutdown condenser is then initiated, the cooldown rate will exceed 2.5°F/min. This cooldown rate exceeds the maximum administrative limit of 60°F/h, as well as the LACBWR Technical Specification limit of 150°F/h averaged over a 1-hour period. To slow the cooldown to the administrative limit, the operator must continue to throttle the shutdown condenser steam inlet valves.

Scoping calculations of the cooldown rate using only the shutdown condenser compared to the cooldown rate at the administrative limit, indicate that the 60°F/h cooldown rate cannot be maintained with the RCS temperature below 337.5°F. Furthermore, with the RCS temperature below 242.5°F, the cooldown rate slows to less than 1°F/h. The quantity of water consumed at the end of 36 hours with a RCS temperature of 235°F is 22,145 gallons.

The LACBWR demineralized water storage tank has a capacity of approximately 30,000 gallons, and therefore, the staff concludes that sufficient demineralized water capacity exists for shutdown condenser shell side water level makeup to conduct a plant cooldown in accordance with BTP RSB 5-1. There is no Technical Specification requiring a minimum water level in the demineralized water storage tank. The backup water supplies to the shutdown condenser originate from the river. Since a shutdown to essentially cold conditions can be performed in a reasonable time, the possible degradation of the condenser tubes from the non-demineralized water is not a safety concern.

The staff, in assuming the only cooling path is via the shutdown condenser, allowed no consideration of the various other modes of decay heat removal that may be available. This assumption permitted the scoping calculation to maximize the postulated amount of demineralized makeup water that would be required for cooldown, but severely limits the predicted cooldown of the RCS temperatures, scopingly calculated to be 235°F at the end of 36 hours.

The alternate methods of decay heat removal include the decay heat cooling system, the primary purification system, and the seal injection system. These three systems are not included on the minimum list of systems required for safe shutdown since all are of single train design, susceptible to a single active failure and loss of offsite power, and not required for a safe shutdown and cooldown. However, to assume all of these systems would fail coincidentally is limiting and highly improbable, since all are designed to withstand normal RCS operating temperature and pressure. Furthermore, a feed and bleed process using the secondary side of the shutdown condenser could be initiated as the cooldown rate slows and the RCS temperature asymptotically approaches 212°F. This would require an operator to momentarily enter the containment environment to establish the drain path of the shutdown condenser shell side.

The staff concludes that the LACBWR can be cooled down in accordance with the intent of BTP RSB 5-1 by using any one of, or a combination of, the alternate decay heat removal methods noted above as dictated by the plant operating conditions and allowing for the ambient heat losses of the RCS to the containment environment.

Table A-1. Plant-Specific Data and Cooldown Assumptions

Plant-Specific Data

Plant	LACBWR
Power	165 MW_t
Initial RCS Temperature	577.5°F
Heat Removal Capacity*	3.93×10^7 Btu/h
Secondary Makeup Water Temperature	100°F
Stored Sensible Heat	Metal - 6.08×10^4 Btu/°F Water - 9.79×10^4 Btu/°F
Demineralized Makeup Water Onsite	~30,000 gal**

Cooldown Assumptions

1. Reactor trips at t = 0.
2. Decay heat is in accordance with Draft ANS-5.1.
3. Plant remains at hot shutdown for 4 hours prior to cooldown.
4. Mass of water in the reactor vessel is constant.
5. Shutdown condenser inlet steam flowrate is throttled to maintain the administrative cooldown rate.

*Using shutdown condenser only.
**This quantity is not included in plant technical specifications.

Table A-2. Calculation of Safe Shutdown Water Requirements

Plant: LACBWR

Phase I Reactor trip to point at which decay heat generation rate equals heat removal capacity: approx. 0.0 min

Phase II Delay prior to cooldown: 4 hours
Decay heat generated prior to cooldown: 2.92×10^7 Btu
Water expended prior to cooldown: 3320 gal

Phase III Cooldown*:

Time (h)	RCS Temperature (°F)	RCS Pressure (psia)	Decay Heat Generated (Btu)
4	577.5	1300	2.92×10^7
8.0	337.5	114	4.86×10^7
15.5**	242.5	26.1	7.79×10^7
36.0	235	22.9	1.37×10^8

Decay heat rate at t = 36.0 h: 2.55×10^6 Btu/h

Feedwater expended during cooldown to 235°F: 18,825 gal

Total feedwater expended: 22,145 gal

---

*Using shutdown condenser only.
**Cooldown rate using only the shutdown condenser decreases to less than 1°F/h at 15.5 h.