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SYSTEMATIC EVALUATION PROGRAM, TOPIC VI-10.A

TESTING OF REACTOR PROTECTIVE SYSTEM AND ENGINEERED SAFETY FEATURES, LA CROSSE BOILING WATER REACTOR

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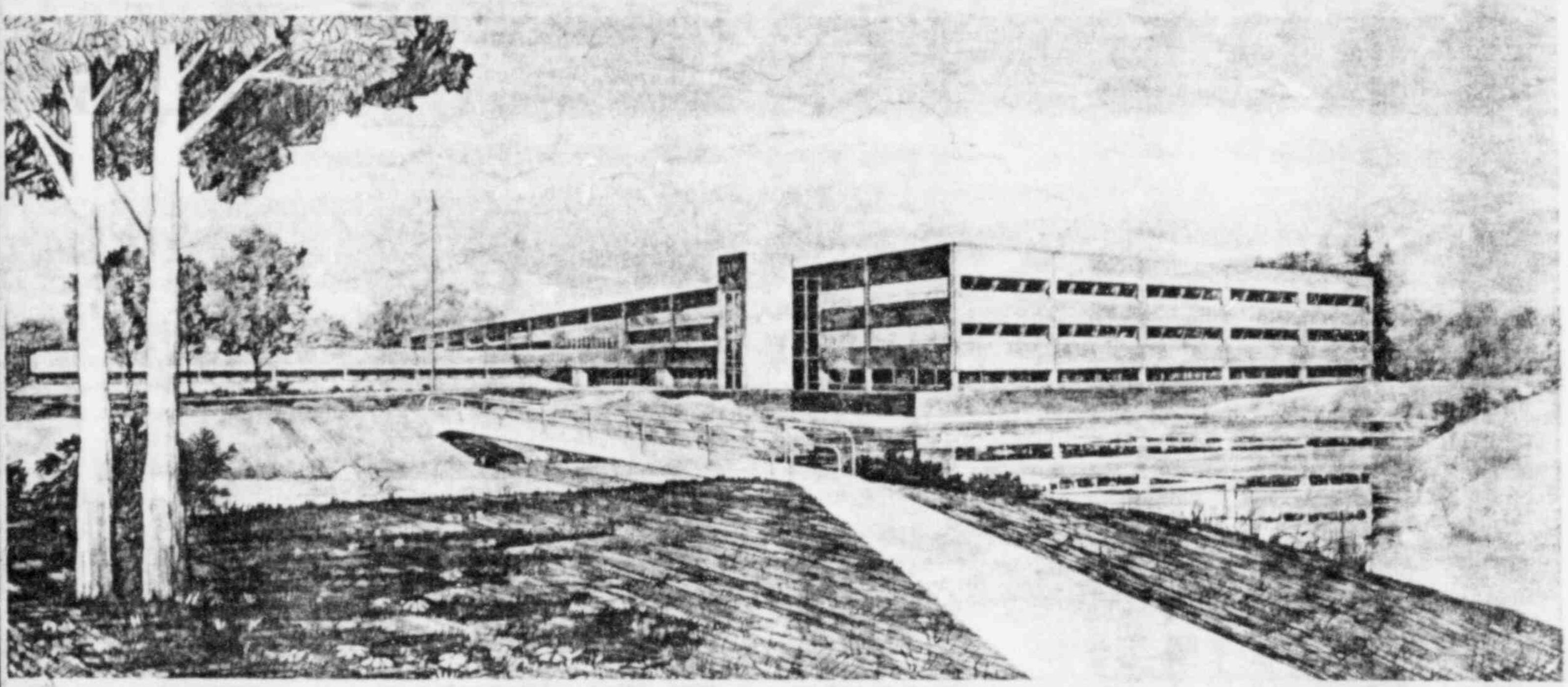

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SYSTEMATIC EVALUATION PROGRAM

TOPIC VI-10.A
TESTING OF REACTOR PROTECTIVE SYSTEM AND
ENGINEERED SAFETY FEATURES

LA CROSSE BOILING WATER REACTOR

Docket No. 50-409

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ABSTRACT

This SEP Technical Evaluation, for the La Crosse Boiling Water Reactor, reviews the scope and frequency of periodic testing of the Reactor Protective System and the Engineered Safety Features and compares the required testing against current licensing criteria.

FOREWORD

This report is supplied as part of the "Electrical, Instrumentation, and Control Systems Support for the Systematic Evaluation Program (II)" being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of Licensing, by EG&G Idaho, Inc., Reliability and Statistics Branch.

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CONTENTS

TABLES

SYSTEMATIC EVALUATION PROGRAM

TOPIC VI - 10.A
TESTING OF REACTOR PROTECTIVE SYSTEM AND ENGINEERED SAFETY FEATURES

LA CROSSE BOILING WATER REACTOR

1.0 INTRODUCTION

The objective of this review is to determine if all reactor protective system (RPS) components, including pumps and valves, are included in component and system tests, if the scope and frequency of periodic testing is adequate, and if the test program meets current licensing criteria. The review will also address these same matters with respect to the engineered safety features (ESF) systems.

2.0 CRITERIA

General Design Criterion 21 (GDC 21), "Protection System Reliability and Testability," states, in part, that:

> The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failure and losses of redundancy that may have occurred.[1]

Regulatory Guide 1.22, "Periodic Testing of the Protection System Actuation Functions," states, in Section D.1.a, that:

> The periodic tests should duplicate, as closely as practicable, the performance that is required of the actuation devices in the event of an accident;

and further, in Section D.4, it states that:

> When actuated equipment is not tested during reactor operation, it should be shown that:
>
> a. There is no practicable system design that would permit operation of the actuated equipment without adversely affecting the safety or operability of the plant,
>
> b. The probability that the protective system will fail to initiate the operation of the actuated equipment is, and can be maintained, acceptably low without testing the actuated equipment during reactor operation, and
>
> c. The actuated equipment can be routinely tested when the reactor is shut down.[2]

IEEE Standard 338-1977, "Periodic Testing of Nuclear Power Generating Station Class 1E Power and Protection Systems," states, in part, in Section 3:

> Overlap testing consists of channel, train, or load-group verification by performing individual tests on the various components and subsystems of the channel, train or load group. The individual component and subsystem tests shall check parts of adjacent subsystems, such that the entire channel, train or load group will be verified by testing of individual components or subsystems.[3]

and, in part, in Section 6.3.4:

> Response time testing shall be required only on safety systems or subsystems to verify that the response times are within the limits of the overall response times given in the Safety Analysis Report.
>
> Sufficient overlap shall be provided to verify overall system response.
>
> The response time shall include as much of each safety system, from sensor input to actuated equipment, as is practicable in a single test. Where the entire set of equipment from sensor to actuated equipment cannot be tested at once, verification of system response time shall be accomplished by measuring the response times of discrete portions of the system and showing that the sum of the response times of all is within the limits of the overall system requirement.

In addition, the following criteria are applicable to the ESF: General Design Criterion 40 (GDC 40), "Testing of Containment Heat Removal System," states that:

> The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure:
>
> a. The structural and leaktight integrity of its components.
>
> b. The operability and performance of the active components of the system.
>
> c. The operability of the system as a whole and under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection systems, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.[4]

GDC 37, "Testing of Emergency Core Cooling System," GDC 43, "Testing of Containment Atmosphere Cleanup Systems," and GDC 46, "Testing of Cooling Water System," are similar.

Standard Review Plan, Section 7.1, Appendix B, "Guidance for Evaluation of Conformance to IEEE STD 279," states, in Section 11, that:

Periodic testing should duplicate, as closely as practical, the overall performance required of the protection system. The test should confirm operability of both the automatic and manual circuitry. The capability should be provided to permit testing during power operation. When this capability can only be achieved by overlapping tests, the test scheme must be such that the tests do, in fact, overlap from one test segment to another.[5]

3.0 REACTOR PROTECTIVE SYSTEM (RPS)

3.1 Description

The Reactor Protection System (RPS) includes the sensors, amplifiers, logic and other equipment essential to the monitoring of selected nuclear power plant conditions. It must reliably effect a rapid shutdown of the reactor if any one or a combination of parameters deviates beyond preselected values to mitigate the consequences of a postulated design basis event.

The RPS parameters and their logic channels, as identified in the La Crosse Technical Specification,[6] are as follows:

PARAMETER:

1. Manual Reactor Trip
2. Source Range, Neutron Flux (Nuclear Channels 1 and 2)
3. Source Range, Reactor Period-Short (Nuclear Channels 1 and 2)
4. Intermediate Range, Neutron Flux (Nuclear Channels 3 and 4)
5. Intermediate Range, Reactor Period-Short (Nuclear Channels 3 and 4)
6. Power Range, Neutron Flux
   a. Reactor Power-High < 15 ± 2% indicated power on Nuclear Channel 7 and 8 (Nuclear Channels 5 and 6)
   b. Reactor Power-High > 15 ± 2% indicated power on Nuclear Channel 7 and 8 (Nuclear Channels 5, 6, 7 and 8 with Automatic Gain Control)
7. Reactor Pressure-High (Pressure Safety Channels 1 and 2)
8. Reactor Power-to-Forced Circulation Flow Abnormal, (Power-Flow Safety Channels 1 and 2)
9. Reactor Coolant Flow Rate-Low (Power-Flow Safety Channels 1 and 2)
10. Reactor Water Level-High (Water Level Safety Channels 1 and 2) and Low Water Level Safety Channels 1, 2 and 3.

11. Main Condenser Vacuum-Low (Main Condenser Vacuum Switches 1 and 2)

12. Main Steam Isolation Valves

    a. Containment Bldg. MSIV Not Fully Open (Valve Closure Relays 1 and 2)

    b. Turbine Bldg. MSIV Not Fully Open (Valve Closure Relays 1 and 2)

    c. Turbine Stop Valve Not Fully Open (Valve Closure Limit Switch)

13. Control Rod Drive Accumulators

    a. Oil Level-Low (Limit Switch)

    b. Gas Pressure-Low (Pressure Switch)

14. Bus Voltages

    a. 2400 v Bus 1A-Low Voltage (Undervoltage Relays Phases A and C)

    b. 2400 v Bus 1B-Low Voltage (Undervoltage Relays Phases A and C)

    c. 2400 v Bus 1A-Low Voltage (Undervoltage Relay Phase A) and 2400 v Bus 1B-Low Voltage (Undervoltage Relay Phase A)

    d. 2400 v Bus 1A-Low Voltage (Undervoltage Relay Phase C) and 2400 v Bus 1B-Low Voltage (Undervoltage Relay Phase C)

    e. Containment Bldg. MCC-1A-Low Voltage (Undervoltage Relays Phases A and C)

    f. Turbine Bldg. MCC-1A Low Voltage (Undervoltage Relays Phases A and C)

15. Reactor Scram Relays

16. Automatic Scram Logic

3.2 Evaluation

Table 1 provides a comparision between the require- ments for surveillance as established by the BWR Standard Technical Specifications (STS) and those set forth by the La Crosse Boiling Water Reactor Technical Specifications.

Evaluation of the La Crosse Technical Specifications indicates that:

1. Six of the La Crosse protective systems comprising the RPS correspond to the BWR Standard Technical Specification RPS; however, the Primary Containment Pressure for La Crosse does not actuate a scram function.

TABLE 1. COMPARISON OF LA CROSSE BOILING WATER REACTOR RPS INSTRUMENT SURVEILLANCE REQUIREMENTS WITH BWR STANDARD TECHNICAL SPECIFICATION REQUIREMENTS (STS)

	CHANNEL CHECK		CHANNEL CALIBRATION		CHANNEL FUNCTIONAL TEST		MODES FOR WHICH SURVEILLANCE IS REQUIRED	
	STS	LA CROSSE	STS[a]	LA CROSSE	STS	LA CROSSE	STS	LA CROSS
1. Intermediate Range Monitors:								
a. Neutron Flux--High Level	D	D[K]	R	---	S/U[b][c], W	S/U[k]	2, 3, 4, 5	---
b. Inoperative	NA	---	NA	---	W	---	2, 3, 4, 5	---
2. Average Power Range Monitor:								
a. Neutron Flux--High, 15%	S	---	S/U[b], W[d]	---	S/U[b][c], W[d]	---	2	---
b. Fixed Biased Neutron Flux--High	S	---	W[e][f], SA	---	S/U[b], W	---	1	---
c. Fixed Neutron Flux--High, 120%	S	---	W[d], SA	---	S/U[b], W	---	1	---
d. Inoperative	NA	---	NA	---	W	---	1, 2, 5	---
e. Downscale	NA	---	NA	---	W	---	1	---
f. LPRM	D	---	(g)	---	NA	---	1, 2, 5	---
3. Reactor Vessel Steam Dome Pressure--High	NA	---	Q	---	M	---	1, 2	---

TABLE 1. (continued)

	CHANNEL CHECK		CHANNEL CALIBRATION		CHANNEL FUNCTIONAL TEST		MODES FOR WHICH SURVEILLANCE IS REQUIRED	
	STS	LA CROSSE	STS[a]	LA CROSSE	STS	LA CROSSE	STS	LA CROSS
4. Reactor Vessel Water Level--Low, Level 3	D	D	Q	R	M	M	1, 2	---
5. Main Steam Line Isolation Valve--Closure	NA	---	R(h)	---	M	S/U, M	1	---
6. Main Steam Line Radiation --High	D	---	R(j)	---	W(i)	---	1, 2	---
7. Primary Containment Pressure--High	NA	---	Q	R	M	---	1, 2	---
8. Scram Discharge Volume Water Level--High	NA	---	R(h)	---	Q	---	1, 2, 5	---
9. Turbine Stop Valve--Closure	NA	---	R(h)	---	M	S/U, M	1	---
10. Turbine Control Valve Fast Closure, Trip Oil Pressure--Low	NA	---	R	---	M	---	1	---
11. Reactor Mode Switch in Shutdown Position	NA	---	NA	---	R	---	1, 2, 3, 4, 5	---
12. Manual Scram	NA	NA	NA	NA	Q	M	1, 2, 3, 4, 5	---
13. Reactor Power-to-Forced Circulation Flow Abnormal	--	D	---	R	---	M	---	---

TABLE 1. (continued)

	CHANNEL CHECK		CHANNEL CALIBRATION		CHANNEL FUNCTIONAL TEST		MODES FOR WHICH SURVEILLANCE IS REQUIRED	
	STS	LA CROSSE	STS[a]	LA CROSSE	STS	LA CROSSE	STS	LA CROSSE
14. Reactor Pressure--High	--	D	---	R	--	S/U, M	---	---
15. Wide Range and Power Range (channels 5, 6, 7, and 8 (k))								
a. Nuclear Instrumentation and Automatic Gain Control Subsystem	--	M	---	---	---	M	---	---
b. Nuclear Instrumentation and Automatic Gain Control Subsystem.	--	M, S	---	---	---	---	---	---
c. Automatic Gain Contol Subsystem	--	M	---	R	---	---	---	---
16. Reactor Water Level High	--	D	---	R	---	M	---	---
17. Reactor Coolant Flowrate Low	--	D	---	R	---	M	---	---
18. Main Condenser Vacuum Low	--	---	---	R	---	S/U, M	---	---
19. Control Rod Drive Accumulators	--	---	---	---	---	W, M	---	---
20. Bus Voltages	--	---	---	---	---	R	---	---

TABLE 1. (continued)

	CHANNEL CHECK		CHANNEL CALIBRATION		CHANNEL FUNCTIONAL TEST		MODES FOR WHICH SURVEILLANCE IS REQUIRED	
	STS	LA CROSSE	STS[a]	LA CROSSE	STS	LA CROSSE	STS	LA CROSSE
21. Reactor Scram Relays	--	---	---	---	---	M	---	---
22. Automatic Scram Logic	--	---	---	---	---	M	---	---

TABLE 1. NOTATION

S - At least once per 12 hours.

D - At least once per 24 hours.

W - At least once per 7 days.

M - At least once per 31 days.

Q - At least once per 92 days.

R - At least once per refueling outage (18 months).

N.A. - Not applicable.

SA - At least once per 6 months.

S/U - Prior to start up.

-- - Not performed or available function.

1. POWER OPERATION.

2. STARTUP.

TABLE 1. (continued)

TABLE 1. NOTATION (continued)

3. HOT SHUTDOWN.

4. COLD SHUTDOWN.

5. REFUELING.

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) Within 24 hours prior to startup, if not performed within the previous 7 days.

(c) The IRM, APRM and SRM channels shall be compared for overlap during each startup, if not performed within the previous 7 days.

(d) When changing from CONDITION 1 to CONDITION 2, perform the required surveillance within 12 hours after entering CONDITION 2.

(e) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. Adjust channel if the absolute difference > 2%.

(f) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(g) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP System.

(h) Physical inspection and actuation of position switches.

(i) Instrument alignment using a standard current source.

(j) Calibration using a standard radiation source.

(k) Testing of the Nuclear Instrumentation and Automatic Gain Control Sub-System shall be done concurrently.

2. The STS requires channel calibration at least once per refueling outage (18 months) for the Intermediate Range Neutron Flux High, the Main Steam Line Isolation Valve Closure, and the Turbine Stop Valve Closure. The La Crosse Technical Specifications do not require channel calibration for these three parameters.

3. The La Crosse Technical Specifications include in the RPS the Main Condenser Vacuum Low, the Control Rod Drive Accumulators, the Bus Voltages, the Reactor Scram Relays, and the Automatic Scram Logic parameters. There is no channel check performed on these parameters. The STS do not require these parameters for the RPS and, therefore, there are no requirements set forth for a channel check. It is left to the NRC Staff to determine whether there is enough plant operating experience to justify no periodic channel check of the above parameters.

4. The La Crosse Technical Specifications do not require a channel calibration for the following parameters.

   Wide Range and Power Range (channels 5, 6, 7, and 8).
   Control Rod Drive Accumulators
   Bus Voltages
   Reactor Scram Relays
   Automatic Scram Logic.

   The STS does not address these parameters; therefore, there are no established requirements for periodic calibration of these parameters. It is noted that these parameters are subjected to periodic channel functional testing. It is left to the NRC Staff to determine whether there is enough plant operation experience to justify no periodic channel calibration and that channel functional testing adequately supports the justification for no periodic channel calibration.

5. There are no requirements specified in the La Crosse Technical Specifications establishing the modes for which surveillance is required.

6. There are no specific testing requirements established in the La Crosse Technical Specifications for the manual or automatic scram. It is assumed that a monthly test of the full scram circuits includes both manual and automatic. However, a test for hot short by means of built-in test switch does not comply with current licensing criteria.

7. There are no specific requirements established in the La Crosse Technical Specifications for response times for those systems comprising the RPS.

4.0 ENGINEERED SAFETY FEATURES SYSTEM

4.1 Description

The Engineered Safety Features System consists of the High Pressure Core Spray, the Alternate Core Spray (Low Pressure Coolant Injection), the Containment Isolation, and the Shutdown Condenser System.

The High Pressure Core Spray (Safety Injection) System is designed to automatically actuate the injection of water from the overhead storage tank into the core spray header which supplies the lines leading to a spray nozzle just above each fuel assembly. Operation of the High Pressure Core Spray (Safety Injection) System is initated automatically by an actuation signal generated upon low reactor water level or high containment building internal pressure. The high pressure core spray pumps can be started and tripped with individual control switches in the control room. However, an interlock prevents manual starting of the pumps unless the reactor is scrammed.

The Low Pressure Coolant Injection System is designed to provide additional cooling water to the core under the conditions of low reactor pressure, low reactor water level, and high reactor building pressure which would exist following the maximum credible accident (MCA). This system also provides the means of flooding the reactor building following an MCA. Approximately 900 gpm of cooling water can be supplied through a nozzle in the top head of the reactor vessel and impinged on perforated deflector plates located above the reactor core. The cooling water is supplied from the river by either of two diesel-driven service water pumps located in the crib house. Two parallel control valves are provided in the line to the reactor and both open on signal, providing redundant paths for coolant injection.

The Containment Isolation System is designed to establish containment integrity within a short time after a major system rupture, preventing escape of fission products from damaged fuel elements to the outside atmosphere. Operation of the Containment Isolation System is initiated automatically by an actuation signal generated upon high containment radiation monitor, high containment building pressure, low water level channels 1 and 2, low steam line pressure and low vacuum, and high reactor pressure. The containment isolation signals can be reset in the control room. All control valves are designed so that initiating the reset signal does not cause automatic reopening of any valve.

The Shutdown Condenser System is designed to condense reactor steam when the reactor is isolated from the main condenser upon closure of the reactor building steam isolation valve or the turbine building steam isolation valve. The system is automatically actuated when (1) the reactor building steam isolation valve is not fully open, (2) the turbine building steam isolation valve is not fully open, or (3) the reactor pressure is above 1325 psig. When the system is initiated automatically, the steam inlet valve and the off-gas vent valve to the waste gas system open immediately.

4.2 Evaluation

Table 2 provides a comparison between the BWR Standard Technical Specification requirements and those of La Crosse Boiling Water Reactor Technical Specifications for the surveillance of the Engineered Safety Features (ESF) System.

Evaluation of the La Crosse Technical Specification indicate that:

1. During each reactor shutdown for major refueling or an interval no greater than one year, the valves for the Containment Isolation System are tested to demonstrate their operability. The Containment Isolation System utilizes the Low Reactor Vessel Water Level parameter which corresponds to that required by the STS and utilizes the High Containment Building Internal Pressure, the High Containment Radiation Monitor, the Low Steam Line Pressure and Low Vacuum, and the High Reactor Pressure parameters which are not required by the STS. The only periodic test of the Low Reactor Vessel Water Level is a channel functional test whereas the STS requires a periodic channel check and channel calibration in addition to the channel functional test. The test frequency for the channel functional test also deviates from that required by the STS.

   The High Containment Building Internal Pressure is periodically subjected to a channel functional test but not subjected to a channel calibration or channel check. The High Containment Radiation Monitor, the Low Steam Line Pressure and Low Vacuum and the High Reactor Pressure parameters are not subjected to periodic channel checks, channel calibrations, and channel functional testing.

   It is left to the NRC Staff to determine whether operating information can justify testing adequacy of the Low Reactor Vessel Water Level and High Containment Building Internal Pressure parameters utilizing channel functional testing only and whether there is enough plant operating experience to justify no testing for the High Containment Radiation Monitor, the Low Steam Line Pressure and Low Vacuum, and the High Reactor Pressure parameters.

2. The Low Pressure Coolant Injection System controls and remotely operated valves are tested semi-annually to demonstrate their operability and an integrated system test is performed annually. The Low Pressure Coolant Injection System utilizes the Low Reactor Vessel Water Level and the Low Reactor Vessel Pressure parameters which correspond to that required by the STS and utilizes the High Reactor Building Pressure parameter which is not required by the STS. The only periodic test of the Low Reactor Vessel Water Level and the Low Reactor Vessel Pressure parameters is a channel functional test, whereas the STS require a periodic channel check and channel calibration in addition to the channel functional test. The test frequency for the channel functional test also deviates from that required by the STS.

TABLE 2. COMPARISON OF LA CROSSE BOILING WATER REACTOR ENGINEERED SAFETY FEATURES (ESF) INSTRUMENT SURVEILLANCE REQUIREMENTS WITH BWR STANDARD TECHNICAL SPECIFICATION (STS) REQUIREMENTS.

	CHANNEL CHECK		CHANNEL CALIBRATION		CHANNEL FUNCTIONAL TEST		MODES FOR WHICH SURVEILLANCE IS REQUIRED	
	STS	LA CROSSE	STS	LA CROSSE	STS	LA CROSSE	STS	LA CROSSE
1. PRIMARY CONTAINMENT ISOLATION								
a. Reactor Vessel Water Level								
(1) Low, Level 3	D	---	M	---	Q	R	1, 2, 3	---
(2) Low Low, Level 2	D	---	M	---	Q	---	1, 2, 3	---
b. Drywell Pressure--High	NA	---	M	---	Q	---	1, 2, 3	---
c. Main Steam Line								
(1) Radiation--High	D	---	W(a)	---	R	---	1, 2, 3	---
(2) Pressure--Low	NA	---	M	---	Q	---	1	---
(3) Flow--High	D	---	M	---	Q	---	1, 2, 3	---
d. Main Steam Line Tunnel Temperature--High	NA	---	M	---	R	---	1, 2, 3	---
e. Condenser Vacuum--Low	NA	---	M	---	Q	---	1, 2#, 3#	---
f. Main Steam Line Tunnel Δ Temperature--High	NA	---	M	---	Q	---	1, 2, 3	---
g. Containment Radiation Monitor--High	---	---	---	---	---	---	---	---

TABLE 2. (continued)

	CHANNEL CHECK		CHANNEL CALIBRATION		CHANNEL FUNCTIONAL TEST		MODES FOR WHICH SURVEILLANCE IS REQUIRED	
	STS	LA CROSSE	STS	LA CROSSE	STS	LA CROSSE	STS	LA CROSSE
h. Low Steam Line Pressure and Low Vacuum	---	---	---	---	---	---	---	---
i. Reactor Pressure High	---	---	---	---	---	---	---	---
2. SECONDARY CONTAINMENT ISOLATION								
a. Plant Exhaust Plenum Radiation--High	D	---	M(a)	---	R	---	1, 2, 3, 5 and *	---
b. Drywell Pressure--High	NA	---	M	---	Q	---	1, 2, 3	---
c. Reactor Vessel Water Level--Low, Level 3	D	---	M	---	Q	---	1, 2, 3	---
d. Refueling Floor Exhaust Radiation--High	D	---	M(a)	---	Q	---	1, 2, 3, 5 and *	---
3. REACTOR WATER CLEANUP SYSTEM ISOLATION								
a. Δ Flow--High	D	---	M	---	R	---	1, 2, 3	---
b. Area Temperature--High	NA	---	M	---	R	---	1, 2, 3	---
c. SLCS Initiation	NA	---	R	---	NA	---	1, 2, 3	---

TABLE 2. (continued)

	CHANNEL CHECK		CHANNEL CALIBRATION		CHANNEL FUNCTIONAL TEST		MODES FOR WHICH SURVEILLANCE IS REQUIRED	
	STS	LA CROSSE	STS	LA CROSSE	STS	LA CROSSE	STS	LA CROSSE
d. Area Ventilation Δ Temperature--High	NA	---	M	---	R	---	1, 2, 3	---
e. Reactor Vessel Water Level--Low Low, Level 2	D	---	M	---	Q	---	1, 2, 3	---
4. <u>REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION</u>								
a. RCIC Steam Line Flow--High	NA	---	M	---	Q	---	1, 2, 3	---
b. RCIC Steam Supply Pressure--Low	NA	---	M	---	Q	---	1, 2, 3	---
c. RCIC Turbine Exhaust Diaphragm Pressure--High	NA	---	M	---	Q	---	1, 2, 3	---
d. RCIC Equipment Room Temperature--High	NA	---	M	---	R	---	1, 2, 3	---
e. RCIC Steam Line Tunnel Temperature--High	NA	---	M	---	R	---	1, 2, 3	---
f. RCIC Steam Line Tunnel Δ Temperature--High	NA	---	M	---	R	---	1, 2, 3	---

TABLE 2. (continued)

	CHANNEL CHECK		CHANNEL CALIBRATION		CHANNEL FUNCTIONAL TEST		MODES FOR WHICH SURVEILLANCE IS REQUIRED	
	STS	LA CROSSE	STS	LA CROSSE	STS	LA CROSSE	STS	LA CROSSE
5. RHR STEAM CONDENSING SYSTEM ISOLATION								
a. RHR Equipment Area Δ Temperature--High	NA	---	M	---	R	---	1, 2, 3	---
b. RHR Area Cooler Temperature--High	NA	---	M	---	R	---	1, 2, 3	---
c. RHR Return Line Flow--High	NA	---	M	---	R	---	1, 2, 3	---
6. SHUTDOWN COOLING SYSTEM ISOLATION								
a. Reactor Vessel Water Level--Low Low, Level 2	D	---	M	---	Q	---	3, 4, 5	---
b. RHR Pump Suction Pressure--High	NA	---	M	---	Q	---	1, 2, 3	---
c. Drywell Pressure--High	NA	---	M	---	Q	---	1, 2, 3	---
d. RHR Pump Suction Flow--High	NA	---	M	---	Q	---	1, 2, 3	---

TABLE 2. (continued)

	CHANNEL CHECK		CHANNEL CALIBRATION		CHANNEL FUNCTIONAL TEST		MODES FOR WHICH SURVEILLANCE IS REQUIRED	
	STS	LA CROSSE	STS	LA CROSSE	STS	LA CROSSE	STS	LA CROSSE
7. LOW PRESSURE CORE SPRAY SYSTEM								
a. Reactor Vessel Water Level--Low Low Low, Level 1	D	---	M	---	Q	SA	1, 2, 3, 4, 5*	---
b. Drywell Pressure--High	NA	---	M	---	Q	---	1, 2, 3	---
c. Injection Valve Differential Pressure --High	D	---	M	---	Q	---	1, 2, 3, 4, 5*	---
d. Pump Discharge Flow-Low (Bypass)	NA	---	M	---	Q	---	1, 2, 3, 4, 5*	---
e. Reactor Vessel Pressure--Low	NA	---	M	---	Q	SA	1, 2, 3, 4, 5*	---
f. Reactor Building Pressure--High	---	---	---	---	---	SA	---	---
8. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM								
a. Reactor Vessel Water Level--Low Low Low, Level 1	D	---	M	---	Q	---	1, 2, 3, 4, 5*	---

	CHANNEL CHECK		CHANNEL CALIBRATION		CHANNEL FUNCTIONAL TEST		MODES FOR WHICH SURVEILLANCE IS REQUIRED	
	STS	LA CROSSE	STS	LA CROSSE	STS	LA CROSSE	STS	LA CROSSE
b. Drywell Pressure --High	NA	---	M	---	Q	---	1, 2, 3	---
c. Injection Valve Differential Pressure --Low	D	---	M	---	Q	---	1, 2, 3, 4, 5*	---
d. RHR Pump Start Time Delay Relay	NA	---	SA	---	R	---	1, 2, 3, 4, 5*	---
e. Pump Discharge Flow-- Low (Bypass)	NA	---	M	---	Q	---	1, 2, 3, 4, 5*	---
9. HIGH PRESSURE CORE SPRAY SYSTEM								
a. Reactor Vessel Water Level--Low Low, Level 2	D	---	M	---	Q	SA	1, 2, 3	---
b. Drywell Pressure--High	NA	---	M	---	Q	---	1, 2, 3	---
c. Condensate Storage Tank Level--Low	NA	---	M	---	Q	---	1, 2, 3	---
d. Suppression Chamber Water Level--High	D	---	M	---	Q	---	1, 2, 3	---

TABLE 2. (continued)

	CHANNEL CHECK		CHANNEL CALIBRATION		CHANNEL FUNCTIONAL TEST		MODES FOR WHICH SURVEILLANCE IS REQUIRED	
	STS	LA CROSSE	STS	LA CROSSE	STS	LA CROSSE	STS	LA CROSSE
e. Reactor Vessel Water Level--High	NA	---	M	---	Q	---	1, 2, 3	---
f. Pump Discharge Pressure--High	NA	---	M	---	Q	---	1, 2, 3	---
g. Pump Suction Pressure --Low	NA	---	M	---	Q	---	1, 2, 3	---
h. HPCS System Flow Rate--High	NA	---	M	---	Q	---	1, 2, 3	---
i. Containment Building Internal Pressure--High	--	---	---	---	--	SA	---	---
10. AUTOMATIC DEPRESSURIZATION SYSTEM								
a. Reactor Vessel Water Level--Low Low Low, Level 1	D	---	M	---	Q	---	1, 2, 3	---
b. Drywell Pressure--High	NA	---	M	---	Q	---	1, 2, 3	---
c. ADS Timer	NA	---	NA	---	R	---	1, 2, 3	---

TABLE 2. (continued)

	CHANNEL CHECK		CHANNEL CALIBRATION		CHANNEL FUNCTIONAL TEST		MODES FOR WHICH SURVEILLANCE IS REQUIRED	
	STS	LA CROSSE	STS	LA CROSSE	STS	LA CROSSE	STS	LA CROSSE
d. Low Pressure Core Spray Pump Discharge Pressure--High	NA	---	M	-	Q	---	1, 2, 3	---
e. RHR LPCI MODE Pump Discharge Pressure--High	NA	---	M	---	Q	---	1, 2, 3	---
f. Reactor Vessel Water Level--Low Low, Level 2	D	---	M	---	Q	---	1, 2, 3	---
11. SHUTDOWN CONDENSER SYSTEM	--		---		--		----	
a. Reactor Building Isolation Valve Not Fully Open	--	---	---	---	--	A	---	---
b. Turbine Building Isolation Valve Not Fully Open	--	---	---	---	--	A	---	---
c. Reactor Pressure Above 1325 psig.	--	---	---	---	--	A	---	---

TABLE 1. NOTATION

S - At least once per 12 hours.

TABLE 2. (continued)

TABLE 1. NOTATION (continued)

D - At least once per 24 hours.

W - At least once per 7 days.

M - At least once per 31 days.

Q - At least once per 92 days.

R - At least once per refueling outage (18 months).

N.A. - Not applicable.

SA - At least once per 6 months.

S/U - Prior to start up.

A - Annually.

-- - Not performed or available function.

1. POWER OPERATION.
2. STARTUP.
3. HOT SHUTDOWN.
4. COLD SHUTDOWN.
5. REFUELING.

When reactor steam pressure $\geq$ (   ) psig and/or any turbine stop valve is open.

TABLE 2. (continued)

TABLE 1. NOTATION (continued)

* When handling irradiated fuel in the secondary containment.

a. Instrument alignment using a standard current source.

b. Not applicable when the system is not required to be OPERABLE.

The High Reactor Building Pressure parameter is periodically subjected to a channel functional test but is not subjected to a channel check or channel calibration.

It is left to the NRC Staff to determine whether there is enough plant operating experience to determine testing adequacy for the Low Pressure Coolant Injection System using only periodic channel functional testing.

3. The High Pressure Core Spray System controls and remotely-operated valves are tested semi-annually to demonstrate their operability and an integrated system test is performed annually. The High Pressure Core Spray System utilizes the Low Reactor Vessel Water Level parameter which corresponds to that required by the STS and utilizes the High Containment Building Internal Pressure parameter which is not required by the STS. The only periodic testing of the Low Reactor Vessel Water Level is a channel functional test whereas the STS require a periodic channel check and channel calibration in addition to the channel functional test. The test frequency for the channel functional test also deviates from that required by the STS.

   The High Containment Building Internal Pressure parameter is periodically subjected to a channel functional test but is not subjected to a channel check or channel calibration.

   It is left to the NRC Staff to determine whether there is enough plant operating experience to determine testing adequacy for the High Pressure Core Spray System using only periodic channel functional testing.

4. The Shutdown Condenser System control valves are tested quarterly to demonstrate their operability and an integrated system test is performed annually. The Shutdown Condenser System is not required by the STS. The system is not subjected to periodic channel checks or channel calibration.

   It is left to the NRC Staff to determine whether there is enough plant operating experience to determine testing adequacy for the Shutdown Condenser System using only annual integrated system testing.

5. There are no requirements established in the La Crosse Technical Specifications determining the modes for which surveillance is required.

6. There are no specific requirements established in the La Crosse Technical Specifications for response times for those systems comprising the ESF.

5.0 SUMMARY

The review of the reference material has determined the present testing and testability of the La Crosse RPS and ESF do not meet the criteria of Section 2.0 of this Technical Evaluation. The Technical Specifications do not establish that testing per the specified criteria of Section 2.0 would adversely affect the safety or the operability of the unit, nor has the licensee established that the probability of system failure is acceptably low without regular testing during reactor operation. The licensee has also not established why the Technical Specifications do not require channel calibration or system time response testing for the RPS and ESF. There is no effort made to determine whether the La Crosse RPS and ESF is adequate for reactor operating purposes. Reference to the Standard Technical Specifications for General Electric Boiling Water Reactors was made to give a general comparison.

6.0 REFERENCES

1. General Design Criterion 21, "Protection System Reliability and Testability," of Appendix A, "General Design Criteria for Nuclear Power Plants," 10 CFR part 50, "Domestic Licensing of Production and Utilization Facilities."

2. Regulatory Guide 1.22, "Periodic Testing of the Protection System Actuation Functions."

3. IEEE Standard 338-1975, "Periodic Testing of Nuclear Power Generating Station Class 1E Power and Protection Systems."

4. General Design Criterion 40, "Testing of Containment Heat Removal Systems," of Appendix A, "General Design Criteria for Nuclear Power Plants," 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

5. Nuclear Regulatory Commission Standard Review Plan, Section 7.1, Appendix B, "Guidance for Evaluation of Conformance to IEEE STD 279."

6. Appendix A to Provisional Operating Authorization No. DPRA-6, "Technical Specifications for the La Crosse Boiling Water Reactor (LACBWR)," Amendment 26, 1981.

7. NUREG-0123, Rev. 1 "Standard Technical Specifications for General Electric Boiling Water Reactors."