September 20, 1982

Docket No. 50-409
LS05-82-09-062

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

Dear Mr. Linder:

SUBJECT: SEP TOPIC VI-10.A, TESTING OF REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES, INCLUDING TIME RESPONSE TESTING - FINAL SAFETY EVALUATION REPORT FOR THE LACROSSE BOILING WATER REACTOR

Enclosure 1 is our contractor's final technical evaluation of this topic for your plant. Enclosure 2 is the staff's final safety evaluation report on this topic. Enclosure 2 is based on Enclosure 1 and the additional information provided in your June 14, 1982 letter.

The staff has concluded that with the exception of the power range instrumentation, an acceptable test and calibration program exists at your plant.

This safety 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 at your facility. This assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this subject are modified before the Integrated Assessment is completed.

SEO4
DSA WSE(38)

Sincerely,

ADD:
G. Staley

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

Enclosures:
As stated

cc w/enclosures:
See next page

* SEE PREVIOUS CONCURRENCE

9/20/82

OFFICE	SEPB *	SEPB *	SEPB *	SEPB *	ORB#5 *	ORB#5 *	AD:SA:DL *
SURNAME	RScholl:bl	TMichaels	RHermann	WRussell	RDudley	DCrutchfield	FMiraglia
DATE	8/19/82	8/19/82	8/20/82	8/26/82	9/10/82	9/13/82	9/13/82

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LS05-82-

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

Dear Mr. Linder:

SUBJECT: SEP TOPIC VI-10.A, TESTING OF REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES, INCLUDING TIME RESPONSE TESTING - FINAL SAFETY EVALUATION REPORT FOR LACROSSE BOILING WATER REACTOR

Enclosure 1 is our contractor's final technical evaluation of this topic for your plant. Enclosure 2 is the staff's final safety evaluation report on this topic. Enclosure 2 is based on Enclosure 1 and the additional information provided in your June 14, 1982 letter.

The staff has concluded that the program for calibration of the power range instrumentation required for the protection of the public health and safety is inadequate at your plant.

This safety 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 at your facility. This assessment may be revised in the future if your facility design is changed or if NRC criteria relating to this subject are modified before the Integrated Assessment is completed.

Sincerely,

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

Enclosures:
As stated

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See next page

OFFICE	SEPB	SEPB	SEPB	SEPB	ORB#5	ORB#5	AD:SA:DL
SURNAME	RScholl:bl	TMichaels	RHermann	NRussell	RDudley	DCrutchfield	TNovak/olito
DATE	8/13/82	8/17/82	8/20/82	8/20/82	9/10/82	9/13/82	9/15/82

Mr. Frank Linder

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SYSTEMATIC EVALUATION PROGRAM
TOPIC VI-10.A
LACROSSE BOILING WATER REACTOR

TOPIC: VI-10.A, Testing of Reactor Trip System and Engineered Safety Features, Including Response Time Testing

I. INTRODUCTION

The purpose of this topic is to review the reactor trip system (RTS) and engineered safety features (ESF) test program for verification of RTS and ESF operability on a periodic basis and to verify RTS and ESF response time in order to assure the operability of the RTS and ESF. Response times should not exceed those assumed in the plant accident analyses. Accordingly, the test program of the RTS and ESF was reviewed in accordance with the Standard Review Plan, including applicable Branch Technical Positions.

II. REVIEW CRITERIA

The review criteria are presented in Section 2 of EG&G Report EGG-EA-5962, "Testing of Reactor Trip System and Engineered Safety Features."

III. RELATED SAFETY TOPICS AND INTERFACES

Topic VI-7.A.3 discusses the question of testing protection systems under conditions as close to design condition as practical. There are no topics that are dependent on the present topic information for their completion.

IV. REVIEW GUIDELINES

Review guidelines are presented in Section 3 of Report EGG-EA-5962.

V. EVALUATION

The LaCrosse design is evaluated against the review criteria in Section 3.2 and 4.2 of the report. The identified differences from current review criteria are as follows:

1. 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 is not performed. The LaCrosse Technical Specifications (pages 45, 46 and 47) do not require channel calibration for these three parameters.

Operating experience at other facilities has shown that the setpoints of bistables and limit switches drift with time. Experience has also shown that neutron detector characteristics change with exposure.

The staff's review of the licensee's test and calibration procedures (as outlined in the overall plant preventive maintenance program) indicates that channel calibration is a part of the inspection and test program.

The Intermediate Range Neutron Flux Monitors are tested annually. The licensee has agreed, by a letter dated June 14, 1982, to perform and document a physical inspection of the valve limit switch mounting at least once per refueling outage in conjunction with the functional testing. Proper switch mounting is required for proper switch actuation.

The turbine stop valve-closure signal feeds the half scram logic, but not the full scram logic. Because no safety credit is taken for the half scram, calibration of the half scram inputs is not required.

2. The LaCrosse 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. Indications derived from independent instrument channels measuring the same identified parameters are not required.

3. The LaCrosse 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 LACBWR preventive maintenance program requires an annual maintenance inspection, and as part of this inspection, the entire neutron monitoring channel is calibrated according to the manufacturers specifications. The channel functional test is performed in such a manner that constitutes a channel calibration. The procedure followed for the functional test is modeled after the manufacturer recommended calibration procedure.

   In addition to the periodic functional test, short and long form calorimetrics are performed to correlate the channel to the heat balance. The short and long form calorimetrics are performed every 8 hours and every 7 days, respectively when the reactor is at power. While operating the four channels are required to indicate within 5% of each other.

Such testing and calibration is sufficient for compensating for random changes in components, but is not sufficient to detect a non-random process such as aging (at the same rate within 5%).

Neutron detector characteristics are known to change with exposure. The licensee checks these instruments by a monthly heat balance. The staff has few details of the heat balance procedure used, and, therefore, cannot conclude that this check can serve as a calibration. An example of a shortcut in doing a heat balance that can affect the accuracy is to assume a feedwater temperature (as is done in the automatic gain adjustment equipment) without regard to the number of feedwater heaters in service.

The failure to calibrate this equipment does not meet the requirements of IEEE Std. 279-1971.

The control rod drive accumulator pressure and level sensors provide partial scram inputs only. Therefore, they are not required for safety and need not be calibrated.

Bus voltage monitors for the 2400 v bus provide signals to both partial and full scram logic. Monitors for the 480 v buses provide full scram inputs. Functional testing, during refueling outages, verifies trip function, minimum voltage and time settings for these relays.

4. For each parameter that is not tested during reactor operation, the licensee should provide the information specified in Position D.4 of R.G. 1.22.

   The licensee's letter of June 14, 1982, provides this information.

   For example it does not appear that the condenser vacuum switches can be tested during operation because the switches share a common line and all of them would have to be isolated from the condenser and bypassed electrically during testing.

   A second example is the bus voltage monitors. These devices historically have a low drift rate and are connected to high voltage circuits. They must be electrically removed from their normal circuits and bypassed to test and this testing can be accomplished when the plant is shutdown. Similar arguments can be made for the main steamline limit switches and containment pressure indicators.

5. There are no requirements established in the LaCrosse Technical Specifications for response times for those systems comprising the RPS. Response time is a design basis for instrumentation. Response time testing is required by IEEE Std. 279-1971.

   Plant test procedures include response time testing of safety systems with large mechanical delays such as the diesel generators and control rod drive systems. Engineering studies such as that submitted with the application for a full-term operating license (Report SS-1178 performed by Gulf Nuclear on Anticipated Transients Without Scram) demonstrated by analysis that the LACBWR fuel will not exceed the critical heat flux in the hot channel or exceed the pressure safety limit of 1540 psig for all transients except in the event of the loss of steam dump capability to the main condenser.

   As a result of this study an operating procedure was written to provide manual trip of the recirculation pumps in the event of an ATWS event which resulted in high reactor vessel pressure conditions and a automatic pump trip system that was placed into service in December 1980.

   The limited risk assessments that were conducted on Palisades and Ginna have shown the staff that additional response time testing of instrumentation and controls does not provide a significant improvement in the likelihood of successful accident mitigation.

6. During each reactor shutdown for major refueling or an interval not 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, High Containment Building Internal Pressure, High Containment Radiation Monitor, Low Steam Line Pressure, and Low Vacuum, and the High Reactor Pressure parameters. A daily channel check and periodic channel calibration should be conducted in addition to the channel functional test. The channel functional tests should also be conducted periodically. The frequency for calibration functional testing, and channel checks should be established in accordance with Section 6.5 and documented in accordance with Section 6.1 of IEEE Std. 338-1975.

   The High Containment Building Internal Pressure is periodically subjected to a channel calibration, but not subjected to a functional test or channel check. The Low Steam Line Pressure parameter is not subjected to periodic channel checks or channel functional testing. These signals are also used to initiate the Low Pressure Coolant Injection System, the High Pressure Core Spray System, and the Shutdown Condenser System.

   All of these instruments are periodically calibrated to satisfy the requirements of IEEE Std. 279-1971 and GDC 21 because they are required for safety. In addition, periodic functional test and channel checks should be performed.

The licensee's letter of June 14, 1982, notes that:

> "The Standard Technical Specifications require a channel calibration quarterly for the reactor scram function. LACBWR has conducted refueling calibrations of the water level channels during the life of the plant. Due to the mechanical sensing line connection and electrical design, calibrating these instrument loops at power would adversely affect the operation of the plant. There are no bypasses on any trip function except the scram, so a calibration at power of one channel would start four diesels, close the MSIV and result in a plant scram.
>
> A scram input logic of 1 out of 3 water levels, the feedwater control water level input, and various containment isolation and emergency system actions cannot be bypassed during reactor operation. Due to the stability of the transmitters and monthly channel functional tests a more frequent calibration is not deemed necessary."

The staff agrees.

Similar calibration arguments are applicable to other reactor parameters but do not appear to be applicable to the radiation monitors. In addition, channel checks are not applicable to pressure switches such as those used to measure vessel level and pressure, condenser vacuum and containment pressure.

However, the licensee has proposed to perform channel checks on the analog channels measuring the same parameters for Containment Building Internal Pressure and Steam Line Pressure to verify that these parameters are not beyond their trip points.

VI. CONCLUSION

As a result of our review, the staff has concluded that, with the exception of the power range instrumentation, an acceptable test and calibration program is in place at LaCrosse.

It is the staff's position that the design of systems which are required for safety shall include provisions for periodic verification that the minimum performance of instruments and control is not less than that which is assumed in the safety analyses. The bases for this position are General Design Criterion 21, Section 3.9 of IEEE Std. 279-1971, and IEEE Std. 388-1971. Therefore, the licensee should implement a program for calibrating the nuclear power range instrumentation in a manner that addresses all important parameters (e.g., feedwater temperature).