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Docket No. 50-255

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Mr. David P. Hoffman
Nuclear Licensing Administrator
Consumers Power Company
212 West Michigan Avenue
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Dear Mr. Hoffman: SEE Repts.

RE: SEP TOPIC III-1, VI-7.A.3, VI-7.B, VI-7.F, VI-10.A, VII-1.A and VII-2 (Palisades Plant)

Enclosed is a copy of the following staff evaluations of Systematic Evaluation Program Topics III-1, Electrical, Instrumentation, and Control Portion of the Classification of Structures, Components and Systems; VI-7.A.3, ECCS Performance-ECCS Actuation System; VI-7.B, Electrical, Instrumentation, and Control Portion of the ESF Switchover from Injection to Recirculation Mode; VI-7.F, Accumulator Isolation Valves Power and Control System Design; VI-10.A, Electrical, Instrumentation, and Control Portions of the Testing of RTS and ESF, Including Response Time; VII-1.A, Electrical, Instrumentation and Control Portions of the Isolation of the RPS from Non-Safety Systems, Including Qualification of Isolation Devices; and VII-2, Electrical, Instrumentation and Control Portions of the ESF System Control Logic and Design for the Palisades Plant.

These assessments compare your facility, as described in Docket No. 50-255, with the criteria currently used by the Regulatory Staff for licensing new facilities. Please inform us if your as-built facility differs from the licensing basis assumed in our assessments within 30 days of receipt of this letter.

These evaluations will be basic input to the integrated safety assessment for your facility unless you identify changes needed to reflect the as-built conditions at your facility. These topic assessments may be revised in the future if your facility design is changed or if NRC criteria relating to this topic is modified before the integrated assessment is completed.

Sincerely,

/S/

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

8012080 748

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

November 12, 1980

Docket No. 50-255
LS05-80-11-011

Mr. David P. Hoffman
Nuclear Licensing Administrator
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

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Sincerely,

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
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SYSTEMATIC EVALUATION PROGRAM
REVIEW OF NRC SAFETY TOPIC 111-1
ASSOCIATED WITH THE ELECTRICAL, INSTRUMENTATION,
AND CONTROL PROTION OF THE CLASSIFICATION
OF STRUCTURES, COMPONENTS, AND SYSTEMS
FOR THE PALISADES NUCLEAR POWER PLANT

DOCKET NO. 50-255

ABSTRACT

This report documents the technical evaluation and review of NRC Safety Topic III-1, associated with the electrical, instrumentation, and control portions of the classification of structures, components, and systems for the Palisades Nuclear Power Plant, using current licensing criteria.

FOREWORD

This report is supplied as part of the Systematic Evaluation Program being conducted for the U.S. Nuclear Regulatory Commission by Lawrence Livermore National Laboratory. The work was performed by EG&G, Inc., Energy Measurements Group, San Ramon Opertions for Lawrence Livermore National Laboratory under U.S. Department of Energy contract number DE-AC08-76NV01183.

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SYSTEMATIC EVALUATION PROGRAM REVIEW OF NRC SAFETY TOPIC III-1
ASSOCIATED WITH THE ELECTRICAL, INSTRUMENTATION, AND CONTROL PORTION
OF THE CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS
FOR THE PALISADES NUCLEAR POWER PLANT

M. W. Nishimura
EG&G, Inc., Energy Measurements Group
San Ramon Operations

1. INTRODUCTION

Some of the SEP plant structures, systems, and components may not be designed to withstand the effects of a safe shutdown earthquake and remain functional. In some cases, systems and components important to safety may not be designed, fabricated, erected, and tested to quality standards commensurate with their safety function.

The compilation of the major systems required for design basis events (DBE) and for safe shutdown of the plant is submitted in support of NRC Safety Topic III-12. This safety topic addresses whether the major systems identified meet current quality standards.

2. REVIEW GUIDELINES

The objective of this review is to identify only the major electrical, instrumentation and control systems (EICS) required for DBE and for safe shutdown of the plant. This identification is to be performed when reviewing each NRC safety topic. A detailed search is not to be made to identify all required systems for DBE and for safe shutdown.

3. COMPILATION OF IDENTIFIED SYSTEMS

3.1 ENGINEERED SAFETY FEATURE SYSTEMS

The following engineered safety feature systems are required for DBE and safe shutdown:

1. Safety injection system
   a. High-pressure safety injection pumps
   b. Low-pressure safety injection pumps
   c. Safety injection tank
2. Containment spray system
3. Containment air coolers
4. Iodine removal system
5. Containment venting charcoal filter
6. Electric hydrogen recombiner system

3.2 REACTOR PROTECTION SYSTEMS

The following reactor protection systems are required for DBE and safe shutdown:

1. Power range safety channels
2. Wide-range logarithmic neutron monitors
3. Reactor coolant flow
4. Thermal margin/low pressurizer pressure
5. High-pressurizer pressure
6. Steam generator level
7. Steam generator pressure

8. Containment pressure
9. Loss of load
10. Protection system logic units
11. Manual trips

3.3 ADDITIONAL SYSTEMS

In addition to the ESF and RPS, the following systems are required for DBE and safe shutdown:

1. Offsite power system
2. Emergency diesel generator
3. Safety injection and refueling water tank
4. Containment sump
5. Control room systems
6. Compressed air system
7. Engineered safeguards local panel-auxiliary building
8. Service water system
9. Component cooling system
10. Auxiliary feedwater system
11. Heating, ventilating and air conditioning systems.

REFERENCES

1. Consumer Power Company, Palisades Final Safety Analysis Report, filmed June 1978.

2. EG&G, San Ramon Division, technical evaluation reports on NRC safety topics:

   a) Safety Topic VI-7.A.3, "ECCS Actuation." July 1980
   b) Safety Topic VI-10.A, "Testing of RTS and ESF, Including Response Time," July 1980.
   c) Safety Topic VII-1.A, "Isolation of RPS from Non-safety Systems, Including Qualification of Isolation Devices," July 1980.
   d) Safety Topic VI-7.B, "ESF Switchover from Injection to Recirculation Mode," July 1980.
   e) Safety Topic VII-2, "ESF System Control Logic and Design," July 1980.

APPENDIX A
NRC SAFETY TOPICS RELATED TO THIS REPORT

1.	Safety Topic VI-7.A.3	"ECCS Actuation."
2.	Safety Topic VI-10.A	"Testing of RTS and ESF, Including Response Time."
3.	Safety Topic VII-1.A	"Isolation of RPS from Non-safety Systems, Including Qualification of Isolation Devices."
4.	Safety Topic VI-7.B	"ESF Switchover from Injection to Recirculation Mode.
5.	Safety Topic VII-2	"ESF System Control Logic and Design."
6.	Safety Topic III-2	"Wind and Tornado Loadings."
7.	Safety Topic III-3	"Hydrodynamic Loads." a) Effects of high water level on structures b) Structural and other consequences (e.g., flooding of safety-related equipment in basements) of failure of underdrain systems c) Inservice inspection of water control structures.
8.	Safety Topic III-4	"Missile Generation and Protection." a) Tornado missiles b) Turbine missiles c) Internally-generated missiles d) Site proximity missiles (including aircraft).
9.	Safety Topic III-5	"Evaluation of Pipe Breaks." a) Effects of pipe break on structures, systems and components inside containment b) Pipe break outside containment.
10.	Safety Topic III-6	"Seismic Design Considerations."

11. Safety Topic III-7 — "Category I Structures Integrity."

a) Inservice inspection, including prestressed concrete containments with either grouted or grouted tendons
b) Design codes, design criteria, load combinations, and reactor cavity design criteria
c) Delamination of prestressed concrete containment structures
d) Containment structural integrity tests.

12. Safety Topic III-8 — "Reactor Vessel Internals Integrity."

a) Loose parts monitoring and core barrel vibration monitoring
b) Control rod drive mechanism integrity
c) Irradiation damage, use of sensitized steel and fatigue resistance
d) Core supports and fuel integrity.

13. Safety Topic III-9 — "Support Integrity."

14. Safety Topic III-10 — "Pumps and Valves Integrity."

a) Thermal-overload protection for motors of motor-operated valves
b) Pump flywheel integrity
c) Surveillance requirementss on BWR recirculation pumps and dischrage valves.

15. Safety Topic III-11 — "Component Integrity."

16. Safety Topic III-12 — "Environmental Qualification of Safety Related Equipment."

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SYSTEMATIC EVALUATION PROGRAM
REVIEW OF NRC SAFETY TOPIC VI-7.A.3, EMERGENCY CORE COOLING SYSTEM PERFORMANCE - ECCS ACTUATION SYSTEM FOR THE PALISADES NUCLEAR POWER PLANT

DOCKET NO. 50-255

ABSTRACT

This report documents the technical evaluation and review of NRC safety topic VI-7.A.3 associated with the electrical, instrumentation and control portions of the emergency core cooling system actuation system for the Palisades Nuclear Power Plant.

FOREWORD

This report is supplied as part of the Systematic Evaluation Program being conducted for the U.S. Nuclear Regulatory Commission by Lawrence Livermore National Laboratory. The work was performed by EG&G, Inc., Energy Measurements Group, San Ramon Operations for Lawrence Livermore National Laboratory under U.S. Department of Energy contract number DE-AC08-76NV01183.

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M. W. Nishimura

EG&G, Inc., Energy Measurements Group
San Ramon Operations

1. INTRODUCTION

This safety topic deals with the testability and operability of the Emergency Core Cooling System (ECCS) actuation system. The ECCS test program should demonstrate a high degree of availability of the system to perform its design function. This report also reviews the plant design to assure that all ECCS components, including the pumps and valves, are included in the component and system test, that the frequency and scope of the periodic testing are adequate, and that the test program meets the requirements of the current licensing criteria detailed in Section 2 of this report.

2. CURRENT LICENSING CRITERIA

GDC 37 [Ref. 1], entitled "Testing of Emergency Core Cooling System," requires that the ECCS be designed to permit appropriate periodic pressure and functional testing to assure the operability of the system as a whole and, under conditions as close to design as practical, to verify the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

Branch Technical Position ICSB 25 [Ref. 2], entitled "Guidance for the Interpretation of GDC 37 for Testing and Operability of the Emergency Core Cooling System as a Whole," states that all ECCS pumps should be included in the system test.

Regulatory Guide 1.22 [Ref. 3], entitled "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.

Standard Review Plan Section 7.3, Appendix A [Ref. 4], entitled "Use of IEEE-Std-279 in the Review of ESFAS and Instrumentation and Controls of Essential Auxiliary Supporting Systems," states in Section 11.b that:

> Periodic testing should duplicate, as closely as practical, the integrated performance required from the ESFAS, ESF systems, and their essential auxiliary supporting systems. If such a "system level" test can be performed only during shutdown, the testing done during power operation must be reviewed in detail. Check that "overlapping" tests do, in

fact, overlap from one test segment to another. For example, closing a circuit breaker with the manual breaker control switch may not be adequate to test the ability of the ESFAS to close the breaker.

Regulatory Guide 1.22 states in section D.4 that:

Where 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 protection 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.

3. REVIEW GUIDELINES

The NRC guidelines used in this review are as follows:

(1) Verify that the test conditions come as close as possible to the actual performance required by ECCS during accident mitigation (GDC 37-item 3, ICSB-25, RG 1.22-D.1.a, SRP 7.3-Appendix A-11.b).

(2) Verify that the system test covers from end to end (sensor through actuated device). If partial tests are performed, verify that the overlapping tests indeed overlap from one test segment to another (GDC 37-item 3, ICSB-25, SRP 7.3 Appendix A-11.b, RG 1.22-D.2).

(3) Summarize the ECCS system surveillance testing interval as defined in the plant's technical specifications.

4. SYSTEM DESCRIPTION

The ECCS, also known as Safety Injection System (SIS), was originally designed to prevent fuel and cladding damage that could interfere with adequate emergency core cooling and to limit the cladding-water reaction to less than approximately 1 percent for all break sizes in the primary system piping, up to and including the double-ended rupture of the largest primary coolant pipe, for any break location, and for the applicable break time. The ECCS also functions to provide rapid injection of large quantities of borated water for added shutdown capability during rapid cooldown of the primary system caused by a rupture of a main steam line.

The ECCS is composed of three subsystems. These subsystems are the high-pressure safety injection system (HPSI), the low-pressure safety injection system (LPSI), and the safety injection tank (SI tank). These subsystems are described as follows:

(1) High-Pressure Safety Injection Pumps

Three high-pressure safety injection (SI) pumps inject borated water at high pressure into the primary coolant system during emergency conditions. The pumps are sized to ensure that, following the rapid depressurization of the primary coolant system and recovering of the core by the SI tanks, one high-pressure pump will keep the core covered with a 25 percent spillage allowance when the recirculation mode starts.

(2) Low-Pressure Safety Injection Pumps

Two low-pressure safety injection (SI) pumps are used to inject large quantities of borated water into the primary coolant system under emergency conditions. They are also used to circulate primary coolant during normal shutdown to remove residual and decay heat.

There are two pumps, each of which can circulate sufficient water to keep the temperature rise through the core to less than the full-power value with the reactor shutdown at the end of core life.

(3) Safety Injection Tanks

Four safety injection (SI) tanks are used to flood the core with borated water following a depressurization of the primary coolant system. The tanks are sized to ensure that three-out-of-four tanks will provide sufficient water to recover the core following a design basis accident (DBA).

5. EVALUATION AND CONCLUSIONS

The following evaluation and conclusions on testability criteria were made for the ECCS and other ESF systems at Palisades Nuclear Power Plant.

Testing of major portions of the ESF control circuits can be accomplished while the plant is at power. More extensive circuit sequence and load testing may be done with the reactor shutdown. The test circuits are designed to test the redundant circuits separately so that the correct operation of each circuit may be verified either by equipment operation or by sequence lights. The test circuit design is such that, should an accident occur while testing is in progress, the test will not interfere with initiation of the safeguards equipment required.

Since the ESF equipment being initiated varies according to whether power is available from the standby source or the diesel generator, a mode selector switch is provided so that either the normal shutdown or the design base accident (DBA) portions of the circuit can be tested separately. Individual momentary type pushbuttons are provided to simulate the SIS in each of the redundant control circuits. The test is in progress only as long as the pushbutton is depressed. Releasing this pushbutton during a test will automatically reset the SIS or DBA sequence relays.

Testing in the "without standby power" mode does not initiate bus load shedding with standby voltage available. After a test, the solenoid-operated valves will reset automatically. Other equipment that is initiated will continue until it is shut down manually.

The system-testing of the safety injection pumps and valves begins when the momentary pushbutton switches are depressed, which simulates the SIS. An alternate method of beginning the test is to trip the two-out-of-four pressurizer low-pressure devices in the initiating circuit matrix at power or shutdown. The bus shedding and the actual sequence loading of the emergency generators can be tested by simulating the loss of standby power.

The system test is considered satisfactory (by the licensee) if control board indication and visual observations indicate that all components have received the safety injection signal in the proper sequence and timing (i.e., the appropriate pump breakers shall have opened and closed, and all valves shall have completed their travel). The test is considered acceptable when the pumps start, reach their rated shutoff heads at minimum recirculation flow, and operate for at least fifteen minutes.

The safety injection pumps, shutdown cooling pumps, and containment spray pumps are started at intervals not to exceed three months. Alternate manual starting between control room console and the C-33 panel are practiced in the test program. During reactor operation, the instrumentation which is necessary to initiate safety injection and containment spray generally is checked daily; the initiating circuits are tested monthly. In addition, the active components (pumps and valves) are tested every three months to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. The test interval of three months is based on the judgment of the licensee that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required), and that more frequent testing would result in increased wear over a long period of time. This report will not conclude whether the test interval of once every three months is adequate. The adequacy of test intervals is discussed in NRC Safety Topic XVI.

The SI tanks are a passive safety feature. In accordance with the specifications, the water volume and pressure in the SI tanks are checked periodically during operation. Each SI tank has two check valves in series between the tank nozzle and the primary coolant system. The pressure control system between the check valves is also used to test the check valves. The check valve nearest the tank may be tested by opening the pressure control valve. As the pressure between the check valves decreases, the valve will open under the influence of tank pressure.

A flowmeter is provided in the test line to measure flow, and indications of tank level and pressure are also available to verify the flow. A line from the discharge header of the charging pumps provides the capability of testing the check valve nearest the primary system.

The pressure control system between the check valves is set for a pressure higher than the primary system pressure. Flow from the charging pump is established and verified by the test-line flowmeter. The pressure between the check valves is gradually increased by increasing the setting on the pressure controller. When the pressure exceeds that existing in the primary coolant system, the check valve will open and the flow from the charging pump will enter the primary coolant system. The lack of flow through the test-line flowmeter will verify that the check valves have opened. These valves will be tested periodically with other components of the system to ensure their operability if needed.

The minimum frequencies for checks, calibrations, and testing of engineered safety feature instrumentation controls are shown in Table 1. This report does not conclude whether the plant complies or does not comply with test frequency criteria. The adequacy of frequency of testing will be discussed in NRC Safety Topic XVI.

Based on the review of the Palisades final safety analysis report [Ref. 5] and technical specifications [Ref. 6], we conclude that the plant complies to current licensing criteria as detailed in Section 2 of this report.

TABLE 1. Minimum frequencies for checks, calibrations and testing of engineered safety feature instrumentation controls.

Channel Description	Surveillance Function	Frequency	Surveillance Method
1. Low-pressure SIS initiation channels	a. Check	S	a. Comparison of four separate pressure indications.
	b. Test[a]	R	b. Signal to meter relay adjust with test device to verify SIS actuation logic.
	c. Test	M[b]	c. Signal to meter relay adjusted with test device.
2. Low-pressure SIS signal Block permissive and auto reset	a. Test[a]	R	a. Part of 1(b).
3. SIS actuation relays	a. Test	Q	a. Simulation of SIS 2/4 logic trip using built-in testing system. Both "standby power" and "no standby power" circuits will be tested for left and right channels. Test will verify functioning of initiation circuits of all equipment normally operated by SIS signals.
	b. Test	R	b. Complete automatic test initiated by same method as Item 1b and including all normal automatic operations.
4. Containment high-pressure channels	a. Calibrate	R	a. Known pressure applied to sensors.
	b. Test	R	b. Simulation of CHP 2/4 logic trip to verify actuation logic for SIS, containment isolation and containment spray
	c. Test	M[b]	c. Pressure switch operation simulated by opening or shorting terminals or pressure applied to the switch.
5. Containment high-radiation Channels	a. Check	D	a. Comparison of four separate radiation level indications.
	b. Calibrate	R	b. Exposure to known external radiation source.

TABLE 1. Minimum frequencies for checks, calibrations and testing of engineered safety feature instrumentation controls (contd).

Channel Description	Surveillance Function	Frequency	Surveillance Method
5. Containment high radiation channels (Contd)	c. Test	M(b)	c. Remote-operated integral radiation check source used to verify instrument operation.
	d. Test	R	d. Simulation of CHR 2/4 logic trip with test switch to verify actuation relays, including containment isolation.
6. Manual SIS initiation	a. Test	R	a. Manual pushbutton test.
7. Manual containment isolation initiation	a. Test	R	a. Manual pushbutton test.
	b. Check	R	b. Observe isolation valves closure.
8. Manual initiation containment spray pumps and valves	a. Test	R	a. Manual switch operation.
9. DBA sequencers	a. Test	Q	a. Proper operation will be verified during SIS actuation test of item 3a.
10. Normal shutdown sequencers	a. Test	h	a. Simulate normal actuation with test-operate switch and verify equipment starting circuits.
11. Diesel start	a. Test	M	a. Manual initiation followed by synchronizing and loading.
	b. Test	R	b. Diesel start, load shed, synchronizing and loading will be verified during item 3b.
	c. Test	P	c. Diesel auto start initiating circuits.
12. SIRW tank-level switch interlocks	a. Test	R	a. Level switches removed from fluid to verify actuation logic.
	b. Test	Q	b. Use SIRW tank control switch to verify actuation of valves.

TABLE 1. Minimum frequencies for checks, calibrations and testing of engineered safety feature instrumentation controls (contd).

Channel Description	Surveillance Function	Frequency	Surveillance Method
13. Safety injection tank-level and pressure instruments	a. Check	S	a. Verify that level and pressure indications is between independent high high/low alarms for level and pressure.
	b. Calibrate	R	b. Known pressure and differential pressure applied to pressure and level sensors.
14. Boric acid tank-level switches	a. Test	R	a. Pump tank below low-level alarm point to verify switch operation.
15. Boric acid heat tracing system	a. Check	D	a. Observe temperature recorders for proper readings.
16. Main steam isolation valve circuits	a. Check	S	a. Compare four independent pressure indications.
	b. Test(3)	R	b. Signal to meter relay adjusted with test device to verify MSIV circuit logic.
17. SIRW tank temperature indication and alarm	a. Check	M	a. Compare independent temperature readouts.
	b. Calibrate	R	b. Known resistance applied to indicating loop.
18. Low-pressure safety injection flow control valve CV-3006.	a. Check	P	a. Observe valve is open with air supply isolated.
19. Safety injection bottle isolation valves	a. Check	P	a. Ensure each valve open by observing valve position indication and valve itself. Then lock open breakers (at MCC-9) and control power (key switch in control room).
20. Safety injection miniflow valves CV-3027, 3056	A. Check	P	a. Verify valves open and HS-3027 and 3056 positioned to maintain them open.

TABLE 1. Minimum frequencies for checks, calibrations and testing of engineered safety feature instrumentation controls (contd).

FREQUENCY ROTATION

Rotation	Frequency
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.
SA	At least once per 6 months.
R	At least once per 18 months.
P	Prior to each startup if not done previous week.
NA	Not applicable.

NOTES:
(a) Calibration of the sensors is performed during calibration of Item 5b, Table 4.4.1.
(b) All monthly tests will be done on only one channel at a time to prevent protection system actuation.
(c) Calibration of the sensors is performed during calibration of Item 7b, Table 4.4.1.

6. SUMMARY

The Palisades Nuclear Power Plant complies to current licensing testing criteria as defined in Section 2 of this report.

REFERENCES

1. Code of Federal Regulations, Title 10, Part 50 (10 CFR 50) Appendix A, (General Design Criteria 37), 1979.

2. U.S. Nuclear Regulatory Commission, Branch Technical Position ICSB 25, "Guidance for Interpretation of GDC 37 for Testing Operability of the Emergency Core Cooling System as a Whole."

3. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.22, "Periodic Testing of the Protection System Actuation Functions."

4. U.S. Nuclear Regulatory Commission, Standard Review Plan, Section 7.3, Appendix A, "Use of IEEE-Std-279 in the Review of the ESFAS and Instrumentation and Controls of Essential Auxiliary Supporting Systems."

5. Consumer Power Company, Palisades Final Safety Analysis Report (filmed June 1978).

6. Consumer Power Company, Palisades Technical Specifications (Date of Issuance 10-16-1972).

APPENDIX A

1. Topic VI-3, "Containment Pressure and Heat Removal Capability."
2. Topic VI-4, "Containment Isolation System."
3. Topic VI-7, "Emergency Core Cooling System."
4. Topic VI-7.C, "ECCS Single Failure Criterion and Requirements for Locking Out Power to Valves Including Independence of Interlocks on ECCS Valves."
5. Topic VI-9, "Main Steam Isolation."
6. Topic VI-10, "Selected ESF Aspects."

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SYSTEMATIC EVALUATION PROGRAM
REVIEW OF NRC SAFETY TOPIC VI-7.B
ASSOCIATED WITH THE ELECTRICAL, INSTRUMENTATION, AND CONTROL PORTIONS OF THE ESF SWITCHOVER FROM INJECTION TO RECIRCULATION MODE
FOR THE PALISADES NUCLEAR POWER PLANT

DOCKET NO. 50-255

ABSTRACT

This report documents the technical evaluation and review of NRC safety topic VI-7.B associated with the electrical, instrumentation, and control portions of the ESF switchover from injection to recirculation mode for the Palisades Nuclear Power Plant, using current licensing criteria.

FOREWORD

This report is supplied as part of the Systematic Evaluation Program being conducted for the U.S. Nuclear Regulatory Commission by Lawrence Livermore National Laboratory. The work was performed by EG&G, Inc., Energy Measurements Group, San Ramon Operations for Lawrence Livermore National Laboratory under U.S. Department of Energy contract number DE-AC08-76NV01183.

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1. INTRODUCTION

Most pressurized water reactors (PWRs) require operator action to realign the emergency core cooling system (ECCS) for the recirculation mode following a loss-of-coolant accident (LOCA). The NRC staff has been requiring (on a case-by-case basis) the use of some automatic features to realign the ECCS from the injection to the recirculation mode of operation. The safety objective of this requirement is to increase the reliability of long-term core cooling by requiring no operator action to change system realignment to the recirculation mode.

This report reviews the ECCS control system and operator action required to align the ECCS from injection mode to recirculation mode following a LOCA.

2. SYSTEM DESCRIPTION

The safety injection and refueling water (SIRW) tank low-level control system is designed to transfer the suction of the safety injection (SI) and containment spray pumps to the containment sump when the SIRW tank is essentially empty. During postaccident cooling of the core, the system performs the functions necessary for recirculating and cooling water which has accumulated in the containment building sump.

The system has redundant low-level control circuits, each of which controls a redundant recirculation loop and the cooling system valves. Each of the redundant control circuits is supplied from a separate preferred a-c source.* Failure of the power source in any one of the level switch circuits will cause the circuit to fail in a mode that initiates recirculation.

In the recirculation mode, the system automatically provides component cooling water to the shell side of the shutdown cooling heat exchangers by opening recirculation valves CV3030 and CV3029 and at the same time closing injection valves CV3057 and CV3031. The circuit is designed on a two-channel concept with each channel initiating the operation of separate and redundant hydraulic loops.

The SIRW tank is provided with four level switches (LS0327 through LS0330) to detect a low level in the tank. Each switch is connected to four auxiliary relays from separate preferred a-c supplies. The

*Failure in any one a-c source of the four level switch circuits will cause that channel to be in a tripped state, thereby changing the logic from two-out-of-four to one-out-of-three logic.

output of these auxiliary relays (contacts LSX0327 through LSX0330) provides two-out-of-four logic matrices for the injection/recirculation valve (CV3030 and CV3031) control relays 4L1 and 4L3. Consistent with the two-channel concept, a separate set of auxiliary relay contacts provides input to injection/recirculation valve (CV3029 and CV3057) control relays 4L2 and 4L4.

Each circuit controls the operation of one of the two redundant component cooling water valves (CV0945 or CV0946) to the shutdown heat exchangers via the component cooling water heat exchangers, as well as the service water valve (CV0823 or CV0826) for service water from one of the component cooling water heat exchangers. The low-level control circuits have no normal or shutdown cooling operating functions, and operate only after the SIRW tank has been nearly emptied.

Coincident two-out-of-four low-level signals initiate the recirculation actuation signal (RAS), which opens the containment sump valves (CV3029 and CV3030), closes the SIRW tank valves (CV3031 and CV3057), stops the low-pressure pumps, and closes the valves in the pump minimum-flow lines. A manual bypass is provided so that the low-pressure injection pumps may be restarted if the operator deems this necessary for long-term core cooling.

The control circuit may be tested while the plant is in operation. This test will initiate the operation of the valves and the trip signal of the low pressure (LP) injection pump. The test may be initiated by the test switches provided in the control room or by actuating the level switches mounted at the SIRW tank. Operation of one of the two redundant test switches on the control panel will deenergize two level switch auxiliary relay circuits and provide a two-out-of-four low-level signal which will initiate operation of the valves. Releasing the test switch will conclude the test, and valve operators will return to the normal positions. In addition, individual valve operation may be tested manually using the valve control switches.

There are check valves on outlets of both loops on the SIRW tank and containment sump. This prevents inadvertent flow of water directly from the SIRW tank to the containment sump; similarly, water flow is inhibited in the reverse direction.

3. EVALUATION AND CONCLUSIONS

The injection and recirculation paths have two emergency cooling-water loops that receive water from the SIRW tank during the injection mode and from the containment sump during the recirculation mode. The d-c power supply to the control logic relays (as shown in Palisades drawing E-246 [Ref. 1]) has two separate d-c power supply sources for each of the two cooling loops.

The two d-c power supplies come from d-c panels D11 and D21. If one d-c power supply fails during the injection phase, the cooling loop for which the power supply has failed will not complete the automatic switch-over to the recirculation mode. The injection/recirculation pump in the failed loop will continue to operate after the SIRW tank water has been depleted. Continued operation may result in pump failure because when d-c power to the control logic relays is lost, the containment sump valve will fail in the closed position and the SIRW tank outlet valve will fail in the open position.

Although damage to one emergency cooling path may occur, single failure criterion is not jeopardized. The second redundant cooling path (independent of the first cooling path) will provide adequate emergency cooling water to the core.

REFERENCES

1. Consumer Power Company drawing numbers M-204 (Piping and Instrumented Diagram-Safety Injection, Containment Spray and Shutdown Cooling System); E-246 (Schematic Diagram SIRW Tank and Containment Sump Valves); E-207 (Schematic Diagram Containment High Pressure, High Radiation, and SIRW Tank Low Level).

2. Consumer Power Company, Palisades Final Safety Analysis Report (filmed June 1978).

APPENDIX A

NRC SAFETY TOPICS RELATED TO THIS REPORT

1.	Safety Topic VI-3	"Containment Pressure and Heat Removal Capability."
2.	Safety Topic VI-4	"Containment Isolation System."
3.	Safety Topic VI-7	"Emergency Core Cooling System."
4.	Safety Topic VI-7.C	"ECCS Single Failure Criterion and Requirements for Locking Out Power to Valves Including Independence of Interlocks on ECCS Valves."
5.	Safety Topic VI-9	"Main Steam Isolation."
6.	Safety Topic VI-10	"Selected ESF Aspects."

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SEP TECHNICAL EVALUATION

TOPIC VI-7.F
ACCUMULATOR ISOLATION VALVES
POWER AND CONTROL SYSTEM DESIGN

PALISADES

Docket No. 50-255

SEP TECHNICAL EVALUATION

TOPIC VI-7.F
ACCUMULATOR ISOLATION VALVES
POWER AND CONTROL SYSTEM DESIGN

PALISADES

1.0 INTRODUCTION

The objective of this review is to determine if the accumulator isolation valve power and control system is in compliance with current licensing criteria.

The specific requirements for accumulator isolation valve power and control system design derive from IEEE 279-1971, which states that the bypass of a protective function will be removed automatically whenever permissive conditions are not met and which also assures that a single electrical failure or operator error will not result in loss of capability of the accumulator to perform its safety function.[1] The criteria are further defined in Branch Technical Positions ICSB 4[2] and ICSB 18[3].

2.0 CRITERIA

Current licensing criteria from ICSB 4 are:

1. Automatic opening of the valves when either primary coolant system pressure exceeds a preselected value (to be specified in the Technical Specifications), or a safety injection signal is present. Both primary coolant system pressure and safety injection signals should be provided to the valve operator.

2. Visual indication in the control room of the open or closed status of the valve.

3. An audible and visual alarm, independent of item 2. above, that is actuated by a sensor on the valve when the valve is not in the fully-open position.

4. Utilization of a safety injection signal to remove automatically (override) any bypass feature that

may be provided to allow an isolation valve to be closed for short periods of time when the reactor coolant system is at pressure (in accordance with provisions of the Technical Specifications).

Current licensing criteria from ICSB 18 are:

1. Failures in both the "fail to function" sense and the "undesirable function" sense of components in electrical systems including valves and other fluid system components should be considered in designing against a single failure, even though the valve or other fluid system component may not be called upon to function in a given safety operational sequence.

2. Where it is determined that failure of an electrical system component can cause undesired mechanical motion of a valve or other fluid system component and this motion results in loss of the system safety function, it is acceptable, in lieu of design changes that also may be acceptable, to disconnect power to the electric systems of the valve or other fluid system component. The plant Technical Specifications should include a list of all electrically-operated valves, and the required positions of these valves, to which the requirement for removal of electric power is applied in order to satisfy the single failure criterion.

3. Electrically-operated valves that are classified as "active" valves , i.e., are required to open or close in various safety system operational sequences, but are manually-controlled, should be operated from the main control room. Such valves may not be included among those valves from which power is removed in order to meet the single failure criterion unless (a) electrical power can be restored to the valves from the main control room, (b) valve operation is not necessary for at least ten minutes following occurrence of the event requiring such operation, and (c) it is demonstrated that there is reasonable assurance that all necessary operator actions will be performed within the time shown to be adequate by the analysis. The plant Technical Specifications should include a list of the required positions of manually-controlled, electrically-operated valves and should identify those valves to which the requirement for removal of electric power is applied in order to satisfy the single failure criterion.

4. When the single failure criterion is satisfied by removal of electrical power from valves described in 2. and 3. above, these valves should have redundant position indication in the main control room and the position indication system should, itself, meet the single failure criterion.

5. The phrase, "electrically-operated valves," includes both valves operated directly by an electrical device (e.g., a motor-operated valve or a solenoid-operated valve) and those valves operated indirectly by an electrical device (e.g., an air-operated valve whose air supply is controlled by an electrical solenoid valve).

3.0 DISCUSSION AND EVALUATION

3.1 Discussion. The Palisades plant uses four Safety Injection (Accumulator) Tanks, each of which has a motor-operated isolation valve.[4] These valves are MO 3041, MO 3045, MO 3049, and MO 3052. Each valve has a single positon indication. The Palisades Technical Specifications require that, prior to attaining reactor criticality, the Safety Injection Tank Isolation Valves must be opened and that power to the valve motors must be removed; method of removal (open breaker, rack-out breaker, or disconnect motor power cables) is not specified.[5] Removal of valve motor power does not disable valve position indication, which is powered from a separate 125 V DC bus.[6] The breakers which supply the valve motors are located inside the containment.[7] The valves are specified by function rather than by valve number. The Technical Specifications also allow any one SI tank to be out of service for no more than one hour during power operation without going to hot shutdown.

3.2 Evaluation. The Palisades accumulator isolation valve power and control system design meets the requirement of ICSB 18, part 2, with the exception that plant Technical Specifications do not list the isolation valves by number. The design does not, however, meet the requirement of ICSB 18, part 4, which must be complied with when removal of valve motor power is used to meet the single failure criterion; only one position indication per valve is available in the control room, a scheme which is inherently single-failure prone. Also, location of the valve motor breakers inside containment poses problems in restoring accumulator isolation capability if necessary.

4.0 SUMMARY

The Palisades accumulator isolation valve power and control system design does not comply with current licensing criteria because (a) plant Technical Specifications do not specify by valve number which valves must be opened and deenergized, (b) control room valve position indication is neither redundant nor single-failure free, and (c) valve motor breakers are located inside containment.

5.0 REFERENCES

1. IEEE Standard 279, "Criteria for Protection Systems for Nuclear Power Generating Stations."

2. Branch Technical Position ICSB 4, "Requirements of Motor-Operated Valves in the ECCS Accumulator Lines."

3. Branch Technical Position ICSB 18, "Application of the Single Failure Criterion to Manually-Controlled Electrically-Operated Valves."

4. Bechtel-Palisades Drawing 5935 M-203, Revision 12, dated 1-14-77.

5. "Technical Specifications for the Palisades Plant," Amendment 31, dated 11-1-77, paragraph 3.3.1.i.

6. Bechtel-Palisades Drawing E-243, sheet 2, No Revision, dated 4-3-79.

7. Bechtel-Palisades Drawing M-5, Sheet 5, Revision 5, dated 10-22-74.

SYSTEMATIC EVALUATION PROGRAM

REVIEW OF NRC SAFETY TOPIC VI-10.A

ASSOCIATED WITH THE ELECTRICAL, INSTRUMENTATION,

AND CONTROL PROTIONS OF THE TESTING OF

RTS AND ESF, INCLUDING RESPONSE TIME

FOR THE PALISADES NCUEYAR POWER PLANT

DOCKET NO. 50-255

ABSTRACT

This report documents the technical evaluation and review of NRC safety topic VI-10.A, associated with the electrical, instrumentation, and control portions of the testing of the RTS and ESF for the Palisades Nuclear Power Plant, using current licensing criteria.

FOREWORD

This report is supplied as part of the Systematic Evaluation Program being conducted for the U.S. Nuclear Regulatory Commission by Lawrence Livermore National Laboratory. The work was performed by EG&G, Inc., Energy Measurements Group, San Ramon Operations for Lawrence Livermore National Laboratory under U.S. Department of Energy contract number DE-AC08-76NV01183.

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1. INTRODUCTION

Safety topic VI-10.A deals with the testability and operability of the Reactor Trip System (RTS) and the Engineered Safety Feature (ESF) systems. The RTS and ESF test program should demonstrate a high degree of availability of these systems and that the response times assumed in the accident analysis are within the design specifications.

The objective of this review is to evaluate the plant design to assure that all RTS components are included in the component and system test, that the frequency and scope of the periodic testing is adequate, and that the test program meets the requirements of the General Design Criteria (GDC) [Ref. 1] and the Regulatory Guides (RG) [Ref. 2,3] as defined in Section 2 of this report.

This report also addresses the containment spray system as an example that is typical to all ESF systems. A review of the plant design will be made to assure that all containment spray system portions of the ESF components, including the pumps and valves, are included in the component and system test, that the frequency and scope of the periodic testing is adequate, and that the test program meets the requirements of the GDC and the criteria of the RGs defined in Section 4 of this report.

2. CURRENT LICENSING CRITERIA

2.1 LICENSING CRITERIA FOR THE REACTOR TRIP SYSTEM (RTS)

GDC 21 [Ref. 1], entitled, "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 failures and losses of redundancy that may have occurred.

Regulatory Guide 1.22 [Ref. 2], entitled, "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.

Regulatory Guide 1.118 [Ref. 3], entitled, "Periodic Testing of Electric Power and Protection Systems," states in part in Section C-12 that:

> Safety system response time measurements shall be made periodically to verify the overall response time (assumed in the safety analysis of the plant) of all portions of the system from and including the sensor to operation of the actuator.
>
> The response time test shall include as much of each safety system, from sensor input to actuated equipment, as possible 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 may be accomplished by measuring the response times of discrete portions of the system and showing that the sum of the response times of all portions is equal to or less than the overall system requirement.

IEEE Std-338-1975 [Ref. 4], entitled, "Periodic Testing of Nuclear Power Generating Station Class 1E Power and Protections Systems," states in Section 3 that:

> 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.

Regulatory Guide 1.22 [Ref. 2] states in Section D.4 that:

Where 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 protection 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.

2.2 LICENSING CRITERIA FOR THE ENGINEERED SAFETY FEATURES (ESF)

All criteria listed in Section 2 of this report are applicable to the ESFs. In addition, the following criteria are also applicable.

GDC 40 [Ref. 1], entitled, "Testing of Containment Heat Removal System," states 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.

Standard Review Plan, Section 7.3, Appendix A [Ref. 5], entitled "Use of IEEE-Std-279 in the Review of the ESFAS and Instrumentation and Controls of Essential Auxiliary Supporting Systems," states in Section 11.b that:

> Periodic testing should duplicate, as closely as practical, the integrated performance required from the ESFAS, ESF systems, and their essential auxiliary supporting systems. If such a "system level" test can be performed only during shutdown, the testing done during power operation must be reviewed in detail. Check that "overlapping" tests do, in fact, overlap from one test segment to another. For example, closing a circuit breaker with the manual breaker control switch may not be adequate to test the ability of the ESFAS to close the breaker.

3. REVIEW GUIDELINES

3.1 REVIEW GUIDELINES FOR THE REACTOR TRIP SYSTEM (RTS)

The NRC guidelines used in reviewing the RTS are as follows:

(1) Verify that the test conditions come as close as possible to actual performance required by RTS (GDC-21, RG 1.22-D.1.a).

(2) Verify that the system test covers from end to end (sensor through actuated device). If partial tests are performed, verify that the overlapping tests indeed overlap from one test segment to another (IEEE-Std-338/1975-3).

(3) Summarize the RTS surveillance testing interval as defined in the plant's technical specification.

(4) Verify that the plant performs a response time testing of sensors and that these response times are within the margin used in the plant's accident analysis (RG 1.118-C.12).

(5) Identify the related NRC safety topics in an appendix to the report.

3.2 REVIEW GUIDELINES FOR THE ESF CONTAINMENT SPRAY SYSTEM

The NRC guidelines used in reviewing the ESF containment spray system are as follows:

(1) Verify that the test conditions come as close as possible to the actual performance required by the ESF/Containment Spray System (GDC-21, GDC-40, SRP 7.3 - Appendix A-11.b).

(2) Verify that the system test covers the system from end-to-end (sensor through actuated device). If partial tests are performed, verify that the overlapping tests indeed overlap from one test segment to another (GDC-40, SRP 7.3 Appendix A-11.b).

(3) Summarize the ESF/Containment Spray System surveillance testing interval as defined in the plant's technical specification.

(4) Verify that the plant performs a response time testing of sensors and that these response times are within the margin used in the plant's accident analysis (RG 1.118-C12).

(5) Identify the related NRC safety topic as an appendix to the report.

4. SYSTEM DESCRIPTION

4.1 DESCRIPTION OF THE REACTOR TRIP SYSTEM (RTS)

The reactor protection system (RPS) includes the sensor instrumentation, amplifiers, logic, and other equipment necessary to monitor selected nuclear steam supply system conditions and to reliably effect a rapid reactor shutdown if any one or a combination of conditions deviates from a preselected operating range. The system functions to protect the reactor core.

The four RPS trip paths consist of redundant sensors, bistables, and relays operating through coincidence logic to maintain power to, or remove it from, the control rod drive (CRD) clutches. Four independent and separate measurement channels normally monitor each safety parameter.* Individual channel trips occur when the measurement reaches a preselected value. Two-out-of-four channel trip logic provides trip signals to one-out-of-six matrix logic units, each of which causes a direct trip of the contactors in the a-c supply to the CRD clutch power supplies.

The RPS is composed of 11 subsystems. These subsystems are described in the following paragraphs.

---

*The FSAR [Ref. 6] states that two measuring channels are used to monitor loss of load and high rate-of-change trips. The Palisades Plant Reactor Protection System Common Mode Failure Analysis report, Section 3, page 3-2, defines loss of load as a one-out-of-one measurement channel.

4.1.1 High Rate-of-Change of Power Trip

A reactor trip for a high rate-of-change of reactor power is provided to protect the reactor against an uncontrolled control rod withdrawal while the core is critical but at low power levels. This is an ancitipatory trip which is not required to protect the reactor since the primary trip is the high-power-level trip.

4.1.2 High Power Level Trip

A reactor trip at a high power level (neutron flux) is provided to shut down the reactor when the indicated neutron flux approaches an unsafe value. The high-power trip signals are initiated by two-out-of-four coincidence logic from the four power-range safety channels.

4.1.3 Low Reactor Coolant Flow Trip

A reactor trip is provided to protect the core from a power to flow mismatch. Provisions are made in the RPS to permit operation at reduced power if one or more of the four coolant pumps (four cooling loops) are taken out of service. For this mode of operation, the low-flow trip setpoints and the overpower trip setpoints are simultaneously changed to allowable values for the selected pump condition by a manual switch equipped with channel separation. This provides a positive means of assuring that the more restrictive settings are used. The switch settings are readily visible to the operator. The flow measurement signals are provided by summing the output of the differential pressure transmitters (0122AX through 0122DX) from each of the four cooling loops to provide an indication of total coolant flow through the reactor. A reactor trip is initiated by two-out-of-four coincidence logic from either of the four independent measuring channels when the flow function falls below a preselected value.

4.1.4 High Pressurizer Pressure Trip

A reactor trip for high pressurizer pressure is provided to prevent excessive blowdown of the primary coolant system by relief action through the pressurizer power-operated relief or safety valves. The trip signals are provided by four narrow-range independent pressure transducers measuring the pressurizer pressure. A reactor trip is initiated by two-out-of-four coincidence logic from the four independent measuring channels if the pressurizer pressure exceeds a preset pressure.

4.1.5 Thermal Margin/Low-Pressure Trip

A trip is initiated by a continuously computed function of primary coolant pressure and thermal power to prevent reactor conditions from violating a minimum departure from the nucleate boiling (DNB) ratio. At constant flow, the temperature rise in the reactor is a function of power, so that the variable trip can be affected by the adjustment of a pressure trip setpoint with reactor inlet and outlet coolant temperatures. At partial flow conditions, the changes in coolant temperature are such that the low thermal margin protection is continued with no change required in the pressure setpoint function. The trip signal is initiated by a two-out-of-four coincidence logic from four independent safety channels, and audible and visual pretrip alarms are actuated to provide for annunciation on approach to reactor trip conditions.

4.1.6 Loss-of-Load Reactor Trip

A reactor trip will automatically be initiated after a turbine trip occurs. The reactor trip will be initited when one of two turbine trip relays is energized. The loss-of-load reactor trip is an anticipatory trip which is not required to protect the reactor since the protection is provided by the high primary system pressure trip. The loss-of-load reactor trip is automatically bypassed when three-out-of-four power range safety channels indicate less than 15 percent of full power.

4.1.7 Low Steam Generator Water Levels 1 and 2 Trip

Low steam generator downcomer water levels will cause a loss of heat removal capability from the primary coolant system. A reactor trip signal is initiated by two-out-of-four logic from four independent downcomer-level differential-pressure transmitters on each steam generator.

4.1.8 Low Steam Generator Pressure Levels 1 and 2 Trip

A reactor trip on low steam generator secondary pressure is provided to protect against excessively high steam flow caused by a steam-line break. An abnormally high main steam flow from either steam generator will cause the secondary pressure to drop rapidly. Four pressure transmitters on each steam generator actuate trip units which are connected in a two-out-of-four logic to initiate the reactor protective action if the steam generator pressure drops below a preselected value. Signals from two-out-of-four indicating meter relays from either steam generator will close the main steam isolation valves on both steam generators.

4.1.9 Manual Trip

A manual reactor trip is provided to permit the operator to trip the reactor. Manual actuation of either of two reactor trip pushbutton switches in the main control room causes direct interruption of the a-c power to the d-c power supplies feeding the control rod drive mechanism (CRDM) electromagnetic clutches.

4.1.10 High Containment Pressure Trip

The high containment pressure reactor trip* is in a diverse backup to the thermal margin/low-pressure trip to ensure that the reactor

*The Palisades FSAR (Section 7.2, Reactor Protective System) [Ref. 6] does not include high containment pressure as one of its RPS channels. However, the high containment pressure is included as part of the RPS channels in Section 4 of the Palisades Technical Specification [Ref. 7].

is tripped before the safety injection sequence and containment spray are initiated in the event of a primary system pipe break (e.g., a LOCA). Four independent pressure switches actuate trip units which are connected in a two-out-of-four coincidence logic to initiate the reactor protective action when the containment pressure reaches 5 psig.

4.1.11 Reactor Protection System Logic Units

The RPS channel trip functions are operated by instrument modules. Each trip function has four independent and separate instrument modules. Each module includes three sealed, electromagnetically-actuated, feed relays and associated contacts. The relays in each module are numbered 1, 2, and 3. The No. 1 relay contacts in the Channel A and B modules are connected into a two-out-of-two logic ladder matrix. (The normally open contacts are used for the logic ladders so that the relays are energized and the contacts closed under operating conditions.) The No. 2 and No. 3 relay contacts in the Channel A and B modules are similarly connected into separate logic ladder matrices. The Channel C and D modules are arranged in a similar manner and have a total of six independent logic matrices, which are designated the AB, AC, AD, BC, BD, and CD. These logic matrices represent all of the two-out-of-four combinations possible.

The contacts of the logic matrix relays are channeled into four trip paths. Each logic matrix has four sealed, electromagnetically-actuated, power-feed relays. Each relay has a single-pole, double-throw (SPDT) contact. The contacts from one logic matrix are placed in series with corresponding contacts from the other logic matrices. Each of these paths is the power supply line to a power trip relay which interrupts the power to the CRDM clutches. Deenergizing of any one power trip relay interrupts (opens) one trip path and effects a one-half trip. Deenergizing any two channels causes a full trip.

If one of the trip modules is to be removed for maintenance, the logic matrices may be changed from a two-out-of-four trip to a two-out-of-three trip by the operating of the logic bypass switch. One key-operated switch is provided for each trip parameter. Only one key is provided for the trips for any one variable to ensure that only one of a group of four can be bypassed at one time.

Where the trip is to be allowed only within selected power ranges, a power-dependent signal is supplied to the trip modules. Below 15 percent power, the reactor trip input from a turbine trip is automatically inhibited. The high-power rate-of-change trip is inhibited below 10^{-4} percent power and also above 15 percent power. Each neutron flux measurement channel supplies the automatic inhibit signals to trip in the same channel. Therefore, channel separation is maintained.

The CRDM clutches are separated into two groups. The clutches in each group are supplied in parallel with low-voltage, d-c power by an ungrounded feedline. Two a-c to d-c converters supply each feedline so that if one converter is cut off, it does not cause release of the clutches. The converters on each side are each supplied by a line from a preferred a-c bus to ensure a continued source of power. Each line passes through two interrupters (each actuated by a separate trip path) in series so that although both a-c lines must be deenergized to release the clutches, there are two separate means of interrupting each line. This arrangement provides a means for the testing of the protective system.

4.2 DESCRIPTION OF THE ESF/CONTAINMENT SPRAY SYSTEM

The functions of the containment spray system are to limit the containment building pressure rise and to reduce the leakage of airborne radioactivity by providing a means for cooling the containment atmosphere after the occurrence of a loss-of-coolant accident (LOCA).

Pressure reduction is accomplished by spraying cool, borated water into the containment atmosphere. Heat removal is accomplished by recirculating and cooling the water through the shutdown heat exchangers.

The system is redundant with a containment air recirculation and cooling system which is completely independent and diversely redundant with the containment spray system. The system is sized such that with a 30-second starting time two of the three pumps will limit containment pressure to less than design pressure following a DBA.

The system consists of three half-capacity pumps, two heat exchangers (shutdown heat exchangers), and all necessary piping, instruments, and accessories. The pumps discharge the borated water through the two heat exchangers to a dual set of spray headers and spray nozzles in the containment. The spray headers are supported from the containment roof trusses and the spray nozzles are arranged in the headers to give complete spray coverage of the containment horizontal cross-section area.

The spray system is initiated by a containment high-pressure signal or remote-manual operation from the control room. If offsite power is available, the signal starts all three spray pumps and opens the isolation valves to the dual containment spray headers. If the offsite a-c power sources are not available, the emergency diesel generators are started and the DBA sequencers allow all three spray pumps to start. Two of the pumps are on one 2400-volt bus, while the third is on a second 2400-volt bus. These buses receive power from the normal or standby sources or, upon loss of these sources, each bus is supplied from a separate emergency diesel generator. Two pumps will meet the capacity requirements in the event of a DBA.

5. EVALUATION AND CONCLUSIONS (RTS)

Provisions are made to permit periodic testing of the complete RPS while the reactor is at operating power levels or when it is shut down. These tests cover the trip actions from sensor input to the protective system to the output to the clutch power supplies. The system test does not inhibit the protective function of the system.

During reactor operation, the measuring channels are checked by comparing the outputs of similar channels and cross-checking them with related measurements. The trip units are tested by inserting a voltmeter into the circuit, noting the signal level, and initiating a test input which is also indicated on the voltmeter. This provides the necessary overlap in the testing process, and also enables the test to establish that the trip can be affected within the required tolerances. The test signal is provided by an external test signal generator which is connected to the trip unit at the signal input terminals. With the test signal generator connected, the desired signal is selected and then inserted into the trip unit by depressing the manual test switch. The test circuit permits various rates of change of the signal input to be used. Trip action (opening) of each of the trip unit relays is indicated by individual lights on the front of the trip unit. The pretrip alarm action is indicated by a separate light.

The sets of trip relays at the output of each coincidence logic matrix are tested one at a time. The test circuits in the logic permit only one pair of coincidence matrix logic relays to be tripped while one set of matrix output relays is held. The application of hold power to one set of matrix output relays denies the power source to the other sets. In testing a logic trip set (e.g., AB), a holding current is initiated in the test coils of the logic trip relays by turning the matrix relay trip test switch to "off" and depressing the matrix logic AB test pushbutton switch. Operation of the matrix trip test switch deenergizes a parallel pair of module trip relays. With the ladder-logic relay contacts open, the logic trip relays may be deenergized one at a time by rotating the matrix relay trip test switch to initiate a half-trip. Indicator lights on the trip relay coils and on the d-c power supply and a-c feedlines provide verification that coil operation and half-trip conditions have occurred.

The minimum frequencies for checks, calibrations, and testing of the RPS are shown in Table 1. This report will not conclude whether the plant complies or does not comply with test frequency criteria. The adequacy of frequency of test will be discussed in NRC Safety Topic XIV.

No information regarding plant response-time verification for the RTS was found in the review of the Palisades Final Safety Analysis Report (FSAR) [Ref. 6], Palisades Technical Specifications [Ref. 7], or other docketed materials. Therefore, based upon these documents, no determination can be made to verify whether the Palisades plant complies or does not comply with the response-time verification criterion (Regulatory Guide 1.118, Section C-12). This subject will be considered later during the integrated design base events (DBE) review.

Based on the review of the Palisades FSAR and the plant technical specification, we conclude that the plant complies to the current licensing criteria detailed in Section 2 of this report, except for Regulatory Guide 1.118, Section C-12.

Table 1. Minimum frequencies for checks, calibrations, and testing of reactor protection system[a].

Channel Description	Surveillance Function	Frequency	Surveillance Method
Power range safety channels	a. Check	S	a. Comparison of four-power channel readings.
	b. Check[b]	D	b. Channel adjustment to agree with heat balance calculation. Repeat whenever flux-ΔT power comparator alarms.
	c. Test	M[c]	c. Internal test signal.[d]
	d. Calibrate	R	d. Channel alignment through measurement/adjustment of internal test points.
Wide-range logarithmic neutron monitors	a. Check	S	a. Comparison of both wide-range readings.
	b. Test	P	b. Internal test signal.
Reactor coolant flow	a. Check	S	a. Comparison of four separate total flow indications.
	b. Calibrate	R	b. Known differential pressure applied to sensors.
	c. Test	M[c]	c. Bistable trip tester.[d][e]
Thermal margin/low pressurizer pressure	a. Check:	S	a. Check:
	(1) Temperature input		(1) Comparison of four separate calculated trip pressure setpoint indications.
	(2) Pressure input		(2) Comparison of four pressurizer pressure indications. (Same as 5a.)
	b. Calibrate:	R	b. Calibrate:
	(1) Temperature input		(1) Known resistance substituted for RTD coincident with known pressure input.
	(2) Pressure input		(2) Part of 5b.
	c. Test	M[c]	c. Bistable trip tester.[d][e]
High-pressurizer pressure	a. Check	S	a. Comparison of four separate pressure indications.
	b. Calibrate	R	b. Known pressure applied to sensors.
	c. Test	M[c]	c. Bistable trip tester.[e]

Table 1. Minimum frequencies for checks, calibrations, and testing of reactor protection system[a] (Contd).

Channel Description	Surveillance Function	Frequency	Surveillance Method
Steam generator level	a. Check	S	a. Comparison of four level indications per generator.
	b. Calibrate	R	b. Known differential pressure applied to sensors.
	c. Test	M[c]	c. Bistable trip tester.[e]
Steam generator pressure	a. Check	S	a. Comparisons of four pressure indications per generator.
	b. Calibrate	R	b. Known pressure applied to sensors.
	c. Test	M[c]	c. Bistable trip tester.[e]
Containment pressure	a. Calibrate	R	a. Known pressure applied to sensors.
	b. Test	M[c]	b. Simulate pressure switch action.
Loss of load	a. Test	P	a. Manually trip turbine auto stop oil relays.
Manual trips	a. Test	P	a. Manually test both circuits.
Reactor Protection System logic units	a. Test	M[c]	a. Internal test circuits.

NOTES:

(a) It is not necessary to perform the specified testing during prolonged periods in the refueling shutdown conditions. If this occurs, omitted testing will be performed prior to returning the plant to service.

(b) Adjust the nuclear gain pot on the ΔT cabinet until readout agrees with heat balance calculations.

(c) All monthly tests will be done on only one of four channels at a time to prevent reactor trip.

(d) Trip setting for operating pump combination only. Settings for other than operating pump combinations must be tested during the routine monthly testing performed when shut down and within four hours after resuming operation with a different pump combination if the setting for that combination has not been tested within the previous month.

(e) The bistable trip tester injects a signal into the bistable and provides a precision readout of the trip set point.

Table 1. Minimum frequencies for checks, calibrations, and testing of reactor protection system[a] (Contd).

FREQUENCY NOTATION

Notation	Frequency
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.
SA	At least once per 6 months.
R	At least once per 18 months.
P	Prior to each startup if not done previous week.
NA	Not applicable.

6. EVALUATION AND CONCLUSIONS (ESF/CONTAINMENT SPRAY SYSTEM)

Testing of major portions of the containment spray system control circuits can be accomplished while the plant is at power. More extensive circuit sequence and load testing may be done with the reactor shut down. The test circuits are designed to test the redundant circuits separately so that the correct operation of each circuit may be verified by either equipment operation or by sequence lights. The test circuit design is such that, should an accident occur while testing is in progress, the test will not interfere with initiation of the safeguards equipment required.

The spray pumps and heat exchangers are located outside the containment to permit access for periodic testing and maintenance during normal plant operation. A recirculation line is provided on the discharge of each spray pump for testing purposes by recirculating water back to the safety injection recirculation water (SIRW) tank. The recirculation line is sized to pass the minimum allowable pump flow.

Since the containment spray system equipment being initiated varies according to whether power is available from the standby source or the diesel generator, a mode selector switch is provided so that either the normal shutdown or the design base accident (DBA) portions of the circuit can be tested separately. Individual momentary-type pushbuttons are provided to simulate the SIS in each of the redundant control circuits. The test is in progress only as long as the pushbutton is depressed. Releasing this pushbutton during a test will automatically reset the SIS or DBA sequence relays.

Further details on testing of the ESF are discussed in a Palisades technical evaluation report, NRC Safety Topic VI-7.A.3, written at EG&G, San Ramon, California [Ref. 8]. The minimum frequencies for checks, calibrations and testing of the containment spray system are also detailed.

This report will not conclude whether the plant complies or does not comply with test frequency criteria. The adequacy of frequency of testing will be discussed in NRC Safety Topic XIV.

No information regarding plant response-time verification for the containment spray system was found in the review of the Palisades FSAR [Ref. 6], the Palisades Technical Specification [Ref. 7], or other docketed materials. Therefore, based upon these documents, no determination can be made to verify whether the Palisades plant complies or does not comply with the response-time verification criterion (Regulatory Guide 1.118, Section C-12). This subject will be considered later during the integrated DBE review.

Based on the review of the Palisades FSAR and the plant technical specifications, we conclude that the plant complies to the current licensing criteria detailed in Section 2 of this report, except for Regulatory Guide 1.118, Section C-12.

7. SUMMARY

The Palisades Nuclear Power Plant complies to current licensing criteria for RTS testing, as defined in Section 2 of this report.

The plant also complies to current licensing criteria for ESF (containment spray system) testing, as defined in Section 2 of this report.

Compliance to response time criterion for both the RTS and ESF cannot be determined. This subject should be addressed in the integrated DBE review.

REFERENCES

1. Code of Federal Regulations, Title 10, Part 50 (10 CFR 50) Appendix A, "General Design Criteria."

2. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.22, "Periodic Testing of the Protection System Actuation Functions."

3. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.118, "Periodic Testing of Electric Power and Protection Systems."

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

5. U.S. Nuclear Regulatory Commission, Standard Review Plan, Section 7.3, Appendix A, "Use of IEEE-Std-279 in the Review of the ESFAS and Instrumentation and Controls of Essential Auxiliary Supporting Systems."

6. Consumer Power Company, Palisades Final Safety Analysis Report, filmed June 1978.

7. Consumer Power Company, Palisades Technical Specification, date of issuance Oct. 16, 1972.

8. EG&G Inc., San Ramon Operations, SEP Review of NRC Safety Topic VI-7.A.3 Associated With the EICS portions of the ECCS, July 1980.

APPENDIX A

NRC SAFETY TOPICS RELATED TO THIS REPORT

1. Topic VI-3, "Containment Pressure and Heat Removal Capability."
2. Topic VI-4, "Containment Isolation System."
3. Topic VI-7, "Emergency Core Cooling System."
4. Topic VI-7.C, "ECCS Single Failure Criterion and Requirements for Locking Out Power to Valves Including Independence of Interlocks on ECCS Valves."
5. Topic VI-9, "Main Steam Isolation."
6. Topic VI-10, "Selected ESF Aspects."

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SYSTEMATIC EVALUATION PROGRAM
REVIEW OF NRC SAFETY TOPIC VII-1.A
ASSOCIATED WITH THE ELECTRICAL, INSTRUMENTATION,
AND CONTROL PORTIONS OF THE ISOLATION
OF THE
RPS FROM NON-SAFETY SYSTEMS, INCLUDING
QUALIFICATION OF ISOLATION DEVICES
FOR THE PALISADES NUCLEAR POWER PLANT

DOCKET NO. 50-255

ABSTRACT

This report documents the technical evaluation and review of NRC safety topic VII-1.A, associated with the electrical, instrumentation, and control portions of the isolation of the reactor protection system (RPS) from non-safety systems and the qualification of isolation devices for the Palisades Nuclear Power Plant, using current licensing criteria.

FOREWORD

This report is supplied as part of the Systematic Evaluation Program being conducted for the U.S. Nuclear Regulatory Commission by Lawrence Livermore National Laboratory. The work was performed by EG&G, Inc., Energy Measurements Group, San Ramon Operations for Lawrence Livermore National Laboratory under U.S. Department of Energy contract number DE-AC08-76NV01183.

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SYSTEMATIC EVALUATION PROGRAM REVIEW OF NRC SAFETY TOPIC VII-1.A
ASSOCIATED WITH THE ELECTRICAL, INSTSRUMENTATION, AND CONTROL PORTIONS
OF THE ISOLATION OF THE RPS FROM NON-SAFETY SYSTEMS,
INCLUDING QUALIFICATION OF ISOLATION DEVICES FOR THE
PALISADES NUCLEAR POWER PLANT

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1. INTRODUCTION

Non-safety systems generally receive control signals from the reactor protection system (RPS) sensor current loops. The non-safety sensor circuits are required to have isolation devices to insure electrical independence of the RPS channels. Operating experience has shown that some of the earlier isolation devices or arrangements at operating plants may not meet current licensing criteria. The safety objective is to verify that operating reactors have RPS designs which provide effective qualified isolation of non-safety systems from safety systems to assure that the safety systems will function as required.

This report reviews the RPS EI&C design features at Palisades Nuclear Power Plant to insure that the non-safety systems which are electrically connected to the RPS are properly isolated from the RPS, and that the isolation devices or techniques meet the current licensing criteria detailed in Section 2 of this report. The qualification of safety-related equipment is not within the scope of this report and is discussed in NRC Safety Topic III-12 [Ref. 1] and NUREG-0458 [Ref. 2].

2. CURRENT LICENSING CRITERIA

GDC 24 [Ref. 3], entitled "Separation of Protection and Control Systems," states that:

> The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leave intact a system that satisfies all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

IEEE Std-279-1971 [Ref. 4], entitled "Criteria for Protection Systems for Nuclear Power Generating Stations," states in Section 4.7.2 that:

> The transmission of signals from protection system equipment for control system use shall be through isolation devices which shall be classified as part of the protection system and shall meet all the requirements of this document. No credible failure at the output of an isolation device shall prevent the associated protection system channel from meeting the minimum performance requirements specified in the design bases.
>
> Examples of credible failures include short circuits, open circuits, grounds, and the application of the maximum credible a-c or d-c potential. A failure in an isolation device is evaluated in the same manner as a failure of other equipment in the protection system.

3. REVIEW GUIDELINES

The NRC guidelines used in reviewing the RTS are as follows:

(1) Verify that the signals used for RPS safety functions are isolated from control or non-safety systems. Identify and describe the type of isolation devices employed (GDC-24, IEEE Std-279-1971 Section 4.7.2).

(2) Identify the related NRC safety topics in an appendix to the report.

4. SYSTEM DESCRIPTION

The reactor protection system (RPS) includes the sensor instrumentation, amplifiers, logic, and other equipment necessary to monitor selected nuclear steam supply system conditions and to reliably effect a rapid reactor shutdown if any one or a combination of conditions deviates from a preselected operating range. The system functions to protect the reactor core.

The four RPS trip paths consist of redundant sensors, bistables, and relays operating through coincidence logic to maintain power to, or remove it from, the control rod drive (CRD) clutches. Four independent and separate measurement channels normally monitor each safety parameter. Individual channel trips occur when the measurement reaches a preselected value. Two-out-of-four channel trip logic provides trip signals to one-out-of-six matrix logic units, each of which causes a direct trip of the contactors in the a-c supply to the CRD clutch power supplies.

The RPS is derived from the following inputs:

(1) High rate-of-change of power

(2) High power level

(3) Low reactor coolant flow

(4) High pressurizer pressure

(5) Thermal margin/low-pressure

(6) Loss of load

(7) Low steam generator water levels

(8) Low steam generator pressure levels

(9) Manual trip

(10) High containment pressure.

5. EVALUATION AND CONCLUSIONS

Three basic types of isolation devices are used between the safety circuits and the non-safety devices. These isolations, which are achieved by the optical isolator, thermistor and resistors, and operational amplifier are described in the following paragraphs.

(1) Resistor Isolation

Tennecomp Systems drawing numbers 114-002815 [Ref. 5] show that isolation for the RPS signals is achieved by 1-kΩ resistors. The following RPS input signals have this type of isolation and input to the computer and the data logger:

a. Steam generator pressure (channel A only).
b. Primary coolant flow (channel A only).
c. Steam generator water level (channel A only).
d. Primary coolant outlet temperature (channel A only).
e. Primary coolant inlet temperature (channel A only).
f. Neutron flux safety (channels A through D).

The following RPS output signals have the same isolation device and input to the computer and the data logger:

a. Neutron flux safety (channels A through D).
b. Reactor trip (channels A through D of the thermal margin, steam generator pressure, steam generator water level, primary coolant flow, high flux, clutch power de-energized, and high pressurizer pressure).

Failure, such as shorting out of resistors and other components, is unlikely because the resistor isolation circuitry is placed on a printed circuit board. The signal voltage level used in this RPS circuit is 0 to 10 volts. The physical location of this circuitry could not be determined from a review of the documents listed in the reference section of this report. [Refs. 5 through 14]. This subject should be addressed in the integrated DBE review.

(2) Optical Isolator and Thermistor Isolation

Tennecomp Systems drawing number 161-002812 [Ref. 5] shows that isolation is achieved by the optical isolator (4N35), thermistor, and resistors. The following RPS signals have this type of isolation and input to the computer and the data logger:

a. Neutron flux safety (channels A through D).
b. Reactor trip (channels A through D from thermal margin, steam generator pressure, steam generator water level, reactor coolant flow, high flux, clutch power deenergized, and pressurizer pressure-high).

The optical isolator and the thermistor isolation circuitry are placed on a printed circuit card. The excitation current to the diode side (safety side) of the optical isolator and safety circuit is limited to 6 mA by two 1,000-ohm thermistors and two 3,000-ohm resistors. The current to the safety circuit will be limited to 8 ma even if both thermistors fail short. The 4N35 optical isolator provides 10^{11}-ohm and 3.5 kV isolation between input (safety side) and output (non-safety side). The maximum collector current of the 4N35-output transistor is rated at 100 mA; the maximum forward current of the 4N35-input diode is rated at 60 mA.

(3) Operation Amplifier Isolation

The signals that originate from the power range safety channel drawer assembly and go to the recorder, remote meter, and auxiliary circuits (as shown in Ref. 6, drawing number 2966-E-2821) are isolated by A709C operational amplifiers with 10-Ω resistors at the inverting and noninverting inputs. Combustion Engineering drawing number J147-1121 [Ref. 6] shows that the two resistors tied to the noninverting side of the amplifier input attenuate the signal by 50 percent to ensure that the common-mode voltage limit of the amplifier is not exceeded.

The two diodes tied to the inputs of the amplifier ensure that the maximum differential-mode voltage can only be the forward voltage drop of one diode or about 0.6 V. Additional diodes in the circuit ensure that the common voltage cannot exceed about 7.0 V at either input to the amplifier. The 0.10-A fuse and the two power-rectifier diodes connected to the output of the amplifier protect the amplifier from faults in the cable or at the load.

All Class 1E or safety-related equipment must satisfy the qualifications of Class 1E equipment for nuclear power plants described in Regulatory Guide 1.89 [Ref. 3]. However, the qualification of safety-related equipment is not within the scope of this report and is discussed in NRC Safety Topic III-12 [Ref. 1] and NUREG-0458 [Ref. 2].

Based on the review of the documents shown in the reference section of this report, the isolation devices and methods used by the Palisades plant comply to the current licensing criteria as detailed in section 2 of this report with the following exceptions.

(1) The RPS steam generator pressure (channel B) and the RPS reactor coolant flow (channel A) signals also go to the input of the plant computer. However, these signals are not isolated.

(2) The resistor and operational amplifier isolation may not be adequate to satisfy the current licensing criteria. This subject should be addressed in the integrated DBE review.

6. SUMMARY

Isolation devices are provided in the interconnections between the RPS system and the computer and logging equipment as required by the current licensing criteria detailed in section 2 of this report. The exceptions to this are the RPS steam generator pressure (channel B) and primary coolant flow (channel A) signals. These two signals input to the plant computer, but they are not isolated.

In addition to the noncompliance indicated above, determination must be made as to whether the resistor isolation and operational amplifier isolation are adequate to satisfy the NRC criteria detailed in section 2 of this report. This subject should be addressed in the integrated DBE review.

REFERENCES

1. Nuclear Regulatory Commission, Safety Topic III-12, Environmental Qualifications of Safety Related Equipment.

2. U.S. Nuclear Regulatory Commission, Short-Term Safety Assessment on the Environmental Qualification of Safety-Related Electrical Equipment of SEP Operating Reactors, NUREG-0458, May 1978.

3. U.S. Nuclear Regulatory Commission, Code of Federal Regulations, Title 10, Part 50 (10 CFR 50), Appendix A (General Design Criteria), 1979.

4. Institute of Electrical and Electronics Engineers, IEEE Std-279-1971.

5. Tennecomp System drawing numbers 114-002815, 141-002815, and 161-002812.

6. Combustion Engineering drawing numbers 2966-E-2821, J147-1121, 2966-E-2858, 2966-D-3196, 2966-D-3198, 2966-E-2846, 2966-D-3106.

7. Fischer-Porter drawing number SC30-1545.

8. Consumer Power Company, Palisades Final Safety Analysis Report (filmed June 1978).

9. Consumer Power Company Palisades Plant Reactor Protection System Common Mode Failure Analysis, March 1975.

10. Consumer Power Company, Palisades Technical Specification, (date of issuance October 16, 1972).

11. NRC memorandum to D. L. Ziemann from R. D. Silver, "Summary of Meeting with Consumer Power Company to discuss SEP Topic VII-1.A for Palisades and Big Rock Point Plants," dated March 21, 1979.

12. Bechtel drawing numbers E-59, E-78, E-83, E-84, E-96, E-615.

13. Harlo drawing number E 12551.

14. Consumer Power Company letter (Steven Cashell) to NRC (Richard D. Silver), dated April 18, 1979.

APPENDIX A

SAFETY TOPICS RELATED TO THIS REPORT

1. TOPIC VII "Instrumentation and Controls."
2. TOPIC VII-1 "Reactor Trip Systems." (IEEE-279).
3. TOPIC VII-1 "Trip Uncertainty and Setpoint Analysis Review of Operating Data Base."
4. TOPIC VII-2 "Engineered Safety Features (ESF) System Control Logic and Design."
5. TOPIC VII-3 "Systems Required for Safe Shutdown."
6. TOPIC VII-4 "Effects of Failure in Non-safety Related Systems on Selected Engineered Safety Features.".
7. TOPIC VII-5 "Instruments for Monitoring Radiation and Process Variables During Accidents."
8. TOPIC VII-6 "Frequency Decay."

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SYSTEMATIC EVALUATION PROGRAM
REVIEW OF NRC SAFETY TOPIC VII-2
ASSOCIATED WITH THE ELECTRICAL, INSTRUMENTATION,
AND CONTROL PORTIONS OF THE ESF SYSTEM
CONTROL LOGIC AND DESIGN
FOR THE PALISADES NUCLEAR POWER PLANT

DOCKET NO. 50-255

ABSTRACT

This report documents the technical evaluation and review of NRC Safety Topic VII-2, associated with the electrical, instrumentation, and control portions of the ESF system control logic and design for the Palisades Nuclear Power Plant, using current licensing criteria.

FOREWORD

This report is supplied as part of the Systematic Evaluation Program being conducted for the U.S. Nuclear Regulatory Commission by Lawrence Livermore National Laboratory. The work was performed by EG&G, Inc., Energy Measurements Group, San Ramon Operations for Lawrence Livermore National Laboratory under U.S. Department of Energy contract number DE-AC08-76NV01183.

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SYSTEMATIC EVALUATION PROGRAM REVIEW OF NRC SAFETY TOPIC VII-2 ASSOCIATED WITH THE ELECTRICAL, INSTRUMENTATION, AND CONTROL PORTIONS OF THE ESF SYSTEM CONTROL LOGIC AND DESIGN FOR THE PALISADES NUCLEAR POWER PLANT

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1. INTRODUCTION

The Engineered Safety Features Actuation Systems (ESFAS) of both PWRs and BWRs may have design features that raise questions about the electrical independence of redundant channels and isolation between redundant ESF channels or trains.

Non-safety systems generally receive control signals from the ESF sensor current loops. The non-safety circuits are required to have isolation devices to insure electrical independence from the ESF channels. The safety objective is to verify that operating reactors have ESF designs which provide effective and qualified isolation between ESF channels, and between ESFs and non-safety systems.

This report reviews the ESF EI&C design features at Palisades Nuclear Power Plant to insure that the non-safety systems electrically connected to the ESFs are properly isolated from the ESFs. This report also reviews the plant's ESFs to insure that there is proper isolation between redundant ESF channels or trains and that the isolation devices or techniques meet the current licensing criteria detailed in Section 2 of this report. The qualification of safety-related equipment is not within the scope of this report and is discussed in NRC Safety Topic III-12 [Ref. 1] and NUREG-0458 [Ref. 2].

2. CURRENT LICENSING CRITERIA

GDC 22 [Ref. 3], entitled "Protection System Independence," states that:

> The protection system shall be designed to assure that the effects of natural phenomena and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or that they shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

GDC 24 [Ref. 3], entitled "Separation of Protection and Control Systems," states that:

> The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection system leave intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

IEEE Std-279-1971 [Ref. 4], entitled "Criteria for Protection Systems for Nuclear Power Generating Stations," states in Section 4.7.2 that:

> The transmission of signals from protection system equipment for control system use shall be through isolation devices which shall be classified as part of the protection system and shall meet all the requirements of this document. No credible failure at the output of an isolation device shall prevent the associated protection system channel from meeting the minimum performance requirements specified in the design bases.

Examples of credible failures include short circuits, open circuits, grounds, and the application of the maximum credible a-c or d-c potential. A failure in an isolation device is evaluated in the same manner as a failure of other equipment in the protection system.

3. REVIEW GUIDELINES

The NRC guidelines used in this review are as follows:

(1) Verify that the signals used for ESF functions are isolated from redundant ESF trains or channels. Review the schematic diagrams to assure that the wiring satisfies the functional logic diagrams in the FSAR or its equivalent (GDC 22).

(2) Verify that qualified electrical isolation devices are utilized when redundant ESF trains or channels share safety signals. Identify and describe the type of isolation device employed (GDC 22).

(3) Verify that the safety signals used for ESF functions are isolated from control or non-safety systems. Identify and describe the type of isolation device employed (GDC 24, IEEE Std-279-1971, Seciton 4.7.2).

(4) Verify that the logic does not contain sneak paths that could cause false operation or prevent required action as the result of operation of plant controls.

(5) Identify the related NRC Safety Topics in an appendix to the report.

4. SYSTEM DESCRIPTION

The engineered safeguards controls, which are initiated by the safety injection signal (SIS), consist of equipment that monitors and selects the available power sources, initiates operation of certain load groups, and will initiate containment isolation when required. The system is designed on a two-independent-channels basis with each channel capable of initiating the load groups for safeguards equipment. This design meets the minimum requirements for safeshut down of the reactor and provides all the necessary functions for operating the systems that are associated with the plant's capability to cope with an abnormal event.

The system has redundant circuitry and physical isolation which is necessary so that a single failure within the system will not prevent proper system action when it is required. The system also has test facilities and alarms that alert the operator when certain components trip, malfunction, or are not available or operable. The controls are interlocked to automatically provide the sequence of operations required to initiate engineered safeguards system operation with or without standby power.

Each of the safety injection system's generating parameters (pressurizer pressure low-low or containment pressure high) has four sensors which utilize a two-out-of-four logic to provide reliable operation with a minimum of nuisance tripping. The four sensors are physically isolated, and operation of any two out of four will initiate the appropriate engineered safeguards action. This action is provided by combining the four sensors into a relay matrix which provides a dual-channel initiation signal. Isolation is maintained in the control panels by locating devices in individual groups and by providing barriers between groups. The cables for the two groups are run in separate raceways.

5. EVALUATION AND CONCLUSIONS

Since both the containment high pressure and the pressurizer pressure low trip circuits are basically the same, only one circuit will be reviewed. The review guidelines listed in Section 3 of this report will be applied to the pressurizer pressure low circuit from the sensor through the output of the actuating logic.

Combustion Engineering drawing 2966-D-3106 [Ref. 5] and Bechtel drawing E-84 [Ref. 6] show that sensor PTD102A receives d-c power from the P-0102A power supply (safety circuit "A"). The power supply, in turn, receives a-c power from the preferred panel Y10 (Y20, Y30 and Y40 supply power to safety circuits B, C and D, respectively). The output from the sensor is fed to high pressure trip unit PA-0102AH, thermal margin low pressure trip unit PA-0102AL, and to pressurizer pressure low-low trip unit PIA-0102ALL. The trip units are all within safety circuit "A"; therefore, they do not require isolation. The drawings do not show interconnections with any additional circuits.

The pressurizer pressure low-low trip unit processes the signal from sensor PT-0102A and converts it to relay logic (two-contact closures). The output of the relay logic is designated as PIA-0102ALL. Bechtel drawing E-206 [Ref. 7] shows that one contact feeds into relays XPA1 and XPA2; the other contact feeds into relays XPA3 and XPA4. Bechtel drawing E-209 [Ref. 8] shows that these four relays make up the two-out-of-four logic circuit which actuates the nine safety injection relays (SIS-1 through SIS-8 and SIS-10). Actuation train "A" has all even-numbered SIS relays, train "B" has all odd-numbered relays. Train "A" receives its power from preferred panel Y20; train "B" from panel Y30.

SIS relays 1 through 8 have 6 contacts (outputs) each, and SIS relay 10 has 12 contacts. There are a total of 60 outputs, all of which are isolated from each other and all of which provide actuation signals to all ESF and other equipment requiring an SIS signal for operation.

Based on the review of the Palisades FSAR [Ref. 9] and the drawings specified, we conclude that the plant complies to the current licensing criteria detailed in Section 2 of this report.

6. SUMMARY

Palisades Nuclear Power Plant complies to current licensing criteria for the ESF system control logic and design, as defined in Section 2 of this report.

REFERENCES

1. Nuclear Regulatory Commission, Safety Topic III-12, Environmental Qualifications of Safety Related Equipment.

2. U.S. Nuclear Regulatory Commission, Short-Term Safety Assessment on the Environmental Qualification of Safety-Related Electrical Equipment of SEP Operating Reactors, NUREG-0458, May 1978.

3. U.S. Nuclear Regulatory Commission, Code of Federal Regulations, Title 10, Part 50 (10 CFR 50), Appendix A (General Design Criteria), 1979.

4. Institute of Electrical and Electronics Engineers, IEEE Std-279-1971.

5. Combustion Engineering drawing 2966-D-3106, "Interconnection Diagram Channel P-0102.

6. Bechtel drawing E-84, "Schematic Diagram-Pressurizer Pressure Control and Measurement Channel Instrumentation."

7. Bechtel drawing E-206, "Schematic Diagram-Safety Injection Signal Auxiliary Circuits."

8. Bechtel drawing E-209, "Schematic Diagram-Safety Injection and Sequence Loading Circuit."

9. Consumer Power Company, Palisades Final Safety Analysis Report, filmed in June 1978.

APPENDIX A
NRC SAFETY TOPICS RELATED TO THIS REPORT

1.	Safety Topic VII-1	"Reactor Trip Systems (IEEE-279)." a. Isolation of reactor protection system from non-safety systems, including qualification of isolation devices b. Trip uncertainty and setpoint analysis review of operating data base.
2.	Safety Topic VII-3	"Systems Required for Safe Shutdown."
3.	Safety Topic VII-4	"Effects of Failure in Non-Safety Related Systems on Selected Engineered Safety Features."
4.	Safety Topic VII-5	"Instruments for Monitoring Radiation and Process Variables During Accidents."
5.	Safety Topic VII-6	"Frequency Decay."
6.	Safety Topic VII-7	"Acceptability of Swing Bus Design on BWR-4 Plants."

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