27
1/30/81 T.S.
MASTER

R-1586

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 DRESDEN STATION, UNIT II
NUCLEAR POWER PLANT

Gerald St. Leger-Barter

November 1980

Lawrence
Livermore
Laboratory

This is an informal report intended primarily for internal or limited external distribution. The opinions and conclusions stated are those of the author and may or may not be those of the Laboratory.

This work was supported by the United States Nuclear Regulatory Commission under a Memorandum of Understanding with the United States Department of Energy.

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

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 Dresden Station Unit II nuclear power plant, using current licensing criteria.

DISCLAIMER

[illegible]

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 under U.S. Department of Energy contract number DE-AC08-76NV01183.

TABLE OF CONTENTS

	<u>Page</u>
1. INTRODUCTION	1
2. CURRENT LICENSING CRITERIA	3
3. REVIEW GUIDELINES	5
4. SYSTEM DESCRIPTION	7
4.1 Core Spray Subsystem	7
4.2 Low Pressure Coolant Injection Subsystem	8
4.3 High Pressure Coolant Injection Subsystem.	8
4.4 Automatic Pressure Relief Subsystem.	9
5. EVALUATION AND CONCLUSIONS	11
6. SUMMARY.	19
REFERENCES.	21
APPENDIX A NRC SAFETY TOPICS RELATED TO THIS REPORT.	A-1

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 DRESDEN STATION UNIT II
NUCLEAR POWER PLANT

GERALD ST. LEGER-BARTER

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 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 plant's ESF EI&C design features 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 and NUREG-0458.

2. CURRENT LICENSING CRITERIA

GDC 22, 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, 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, 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 following NRC guidelines were used for this review:

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).

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).

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, Section 4.7.2).

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 control.

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

4. SYSTEM DESCRIPTION

Means are needed to provide continuity of core cooling during those postulated accident conditions where it is assumed that mechanical failures occur in the primary system and coolant is partially or completely lost from the reactor vessel. Under these circumstances core cooling is accomplished by means of the emergency core cooling system (ECCS). The ECCS consists of two independent core spray subsystems, the low pressure coolant injection (LPCI) subsystem, the high pressure coolant injection (HPCI) subsystem and the automatic pressure relief subsystem.

4.1 CORE SPRAY SUBSYSTEM

The core spray subsystem consists of two independent spray systems each with its own pump, valves, and associated piping and instrumentation. The water source is common to both systems and can be from the suppression pool in the torus or, by appropriate valving, from the contaminated demineralized water storage tank.

Initiation of the core spray subsystem occurs on signals indicating reactor low-low water level and reactor low pressure or high drywell pressure. Low-low water level and high drywell pressure are each detected by four independent level and pressure switches connected in a form of one-of-two-twice logic array. Water injection can start when the admission valve is opened and when the reactor vessel pressure drops below pump discharge pressure (350 psig). Rated flow is sprayed over the top of the core at 90 psig in the reactor vessel. Opening of the admission valves is accomplished only after the reactor pressure decays to approximately the design discharge pressure of the pump, at which time the permissive signal to open the valves is initiated by two pressure switches connected in a one-out-of-two logic array.

4.2 LOW PRESSURE COOLANT INJECTION SUBSYSTEM (LPCI)

The LPCI subsystem consists of two main subdivisions: one the LPCI system, and the other the containment cooling system. The major equipment of the entire subsystem consists of two heat exchangers, four containment cooling service water pumps, four main system pumps, two drywell spray headers, a suppression chamber spray header, and associated valving, piping and instrumentation.

The system pumps are activated on either a signal of reactor low-low water level and reactor low pressure or a signal of high drywell pressure similar to that received by the core spray pumps. The initiation signal also trips the recirculation pumps and supplies a start signal to the diesel generator which will provide power for the pump prime mover if normal auxiliary power has failed. The valves in the high pressure part of the system are activated on a preset reactor low pressure signal similar to that of the valving on the core spray subsystem, thereby establishing a flow path.

4.3 HIGH PRESSURE COOLANT INJECTION SUBSYSTEM (HPCI)

The HPCI subsystem consists of a single steam turbine driving a multi-stage high pressure pump and a gear driven single stage booster pump, valves, high pressure piping, water sources, and instrumentation. The turbine is driven with steam from the reactor vessel. Exhaust steam from the turbine is discharged to the suppression pool. Suction for the HPCI pump is taken from the suppression pool.

Initiation of operation of the system is on a signal of either reactor low water level or high drywell pressure. These level and pressure switches are in a one-of-two-twice logic array similar to the reactor protection system. The system will automatically maintain reactor water level between low level and high level if required flow is within the capacity of the pump and the reactor is not depressurized below 165 psia.

4.4 AUTOMATIC PRESSURE RELIEF SUBSYSTEM (ADS)

The automatic pressure relief subsystem is provided for backup of the HPCI subsystem and performs the functions of vessel depressurization for all small breaks. When the automatic pressure relief subsystem is actuated, the critical flow of steam through the relief valves results in a maximum energy removal rate with a corresponding minimum mass loss. Since the automatic pressure relief subsystem does not provide coolant makeup to the reactor, its function is only in conjunction with the LPCI or core spray subsystems as a backup to the HPCI.

Automatic actuation requires coincident indication of reactor water low-low level and drywell high pressure which is maintained for a period of 2 minutes. It also requires LPCI pressure switches to be made up and/or core spray pressure switches to be made up (i.e., their pump outputs to be above preset levels). There are two actuation chains and each circuit requires the same parameters for actuation.

5. EVAULATION AND CONCLUSIONS

The primary sensors for initiation of the ECCS functions were determined to be pressure and level switches. These switches and their associated ESF systems are tabulated in Table 5.1. It is noted that some of the switch functions are shared but that the isolation between functions is by the use of separate contacts from the relay operated by the pressure/level switch. The reference drawings also note the two switch functions are incorporated in the pressure switches operated by a common actuator. The parameters that initiate the primary function are listed but not all parameters which control the valve sequencing are shown as the isolation carried through for them is similar to that for initiation.

The systems were reviewed in accordance with the review guidelines for isolation and any apparent sneak paths for false operations or inhibition of operation. The isolation of each system from other functions is accomplished by use of separate sensor switches and/or separate contacts.

Based on the review of plant drawings it is concluded that the ESF systems are adequately isolated from control or non-safety systems and each other in accordance with the requirements of section 2 of this report.

TABLE 5.1

ECCS ACTUATION SYSTEM PRIMARY SENSOR SWITCHES

<u>FUNCTION</u>	<u>RELAY</u>	<u>RELAY</u>	<u>ECCS LOGIC</u>
-----------------	--------------	--------------	-------------------

Drywell 1 PSI High Pressure <u>P.S.</u>			
---	--	--	--

2202-5 1501-62A	902-32 1530-134		LPCI
--------------------	--------------------	--	------

2202-5 1501-62B	902-33 1530-234		LPCI
--------------------	--------------------	--	------

2202-5 1501-62C	902-32 1530-199		LPCI
--------------------	--------------------	--	------

2202-5 1501-62D	902-33 1530-299		LPCI
--------------------	--------------------	--	------

<u>FUNCTION</u>		<u>RELAY</u>		<u>RELAY</u>	<u>ECCS LOGIC</u>
-----------------	--	--------------	--	--------------	-------------------

REACTOR LOW WATER
LEVEL LIS

<u>FUNCTION</u>	<u>RELAY</u>	<u>RELAY</u>	<u>ECCS LOGIC</u>
<u>CORE SPRAY PUMP</u> <u>DISCHARGE P.S.</u>			
<u>2202-19A</u> <u>1430-1466A</u>	→ <u>902-32</u> <u>1430-125A</u>	→	ADS
<u>2202-19B</u> <u>1430-1466B</u>	→ <u>902-33</u> <u>1430-125B</u>	→	ADS
<u>2202-19A</u> <u>1430-1466C</u>	→ <u>902-32</u> <u>1430-128A</u>	→	ADS
<u>2202-19B</u> <u>1430-1466D</u>	→ <u>902-33</u> <u>1430-128B</u>	→	ADS

FUNCTIONRELAYRELAYECCS LOGICLPCI PUMP
DISCHARGE P.S.

6. SUMMARY

Based on the review of Dresden Station Unit II Plant drawings it is concluded that the isolation of the ESF systems satisfies the current licensing requirements in section 2 of this report.

REFERENCE

1. Code of Federal Regulations, Title 10, Part 50 (10 CFR 50), 1979, Appendix A, (General Design Criteria).
2. Commonwealth Edison Company, Dresden Station Unit II Final Safety Analysis Report.
3. Dresden II Mechanical Drawings: M-25, January 1978; M-26-1, August 1977; M-27, April 1977.
4. Dresden II Electrical Drawings: 12E2429, September 1976; 12E2430, February 1977; 12E2435, February 1977; 12E2436, September 1976; 12E2437, September 1976; 12E2438, September 1976; 12E2461, September 1976; 12E2462, September 1976; 12E2527, September 1976; 12E2528, December 1976; 12E2529, December 1976; 12E2530, December 1976.

APPENDIX A

1. Topic VI - 7.A.3 "Testability and Operability of the ECCS Actuation System".
2. Topic VI - 10.A "Testing of RTS and ESF including Response Time Testing".
3. Topic VI - 10.B "Shared ESFs On-Site Emergency Power and Service Systems for Multiple Unit Facilities".
4. Topic VII - 1.A "Isolation of the RPS from Non-Safety Systems".
5. Topic VII - 4 "Effects of Failure in Non-Safety Related Systems on Selected ESF's".