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# SYSTEMATIC EVALUATION PROGRAM REVIEW OF NRC SAFETY TOPIC VI-10.A ASSOCIATED WITH THE ELECTRICAL, INSTRUMENTATION AND CONTROL PORTIONS OF THE TESTING OF REACTOR TRIP SYSTEM AND ENGINEERED SAFETY FEATURES, INCLUDING RESPONSE TIME FOR THE DRESDEN STATION, UNIT II NUCLEAR POWER PLANT

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## 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 reactor trip systems and engineered safety features including response time for the Dresden II nuclear power plant, using current licensing criteria.

## FOREWORD

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## 1.0 INTRODUCTION

This safety topic 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 the systems and that the response times assumed in the accident analysis are within the design specifications.

This report reviews 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) and the Regulatory Guides (RG) defined in Section 2 of this report.

This report will also address the containment spray system as a typical example 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 RGs defined in Section 4 of this report.

## 2. CURRENT LICENSING CRITERIA

### 2.1 LICENSING CRITERIA FOR THE REACTOR TRIP SYSTEM (RTS)

GDC 21, 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 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 1.22 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.

Regulatory Guide 1.118, entitled "Periodic Testing of Electric Power and Protection Systems", Section C-12 describes in part 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 entitled "Periodic Testing of Nuclear Power Generating Station Class 1E Power and Protection 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.

## 2.2 CURRENT LICENSING CRITERIA OF THE ENGINEERED SAFETY FEATURES (ESF)

All criteria listed in Section 2 of this report are applicable to the engineered safety feature systems. In addition, the following criteria are also applicable.

GDC 40, 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, 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 supporting 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 GUILDELINES (RTS)

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

B. 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).

C. Summarize the RTS surveillance testing interval as defined in the plant's technical specification.

D. 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).

E. Identify the related NRC safety topics in an appendix to the report.

## 3.2 REVIEW GUIDELINES (ESF/CONTAINMENT SPRAY SYSTEM)

A. Verify that the test condition came 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).

B. Verify that the system test covers from the system 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).

C. Summarize the ESF/containment spray system surveillance testing interval as defined in the plant's technical specification.

D. 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).

E. Identify the related NRC safety topic as an appendix to the report.

# 4. SYSTEM DESCRIPTIONS

## 4.1 SYSTEM DESCRIPTION (REACTOR PROTECTION SYSTEM)

The reactor protection system (RPS) receives signals from plant instrumentation indicating the approach of an unsafe operating condition, actuates alarms, prevents control rod motion, and initiates load cutback, and/or opens the reactor trip breakers depending upon the severity of the condition.

The Reactor Protection System is designed to:

1. Prevent, in conjunction with the containment and containment isolation system, the release of radioactive materials in excess of the limitations of 10CFR100 as a consequence of any of the design basis accidents.

2. Prevent fuel damage following any single equipment malfunction or single operator error.

3. Function independently of other plant controls and instrumentation.

4. Function safely following any single component malfunction.

In order to meet its design requirement, the reactor protection system, under various conditions, initiates a reactor scram. The reactor protection system is referred to sometimes as the dual logic reactor protection system and has been utilized on most General Electric reactor plants.

This part of the report is concerned with the reactor trip system (RTS) portion of the RPS and the licensing criteria will be applied only to the RTS here.

The system is made up of two independent logic channels, each having two subchannels of tripping devices. Each subchannel has an input from at least one independent sensor, monitoring each of the critical parameters.

The output of each pair of subchannels is combined in a one-out-of-two logic: That is, an input in either one or both of the independent subchannels will produce a logic channel trip. Both of the other two subchannels are likewise combined in a one-out-of-two logic, independent of the first logic channel. The outputs of the two logic channels are combined in two-of-two arrangement so that they must be in agreement to initiate a scram. An off-limit signal in one of the subchannels in one of the logic channels must be confirmed by any other off-limit signal in one of the subchannels of the remaining logic channel to provide a scram.

Theoretically, this system's reliability is slightly higher than that of a 2-out-of-3 system and slightly lower than that of a 1-out-of-2 system. However, since the differences are slight, they can, in a practical sense, be neglected. The advantage of the dual logic channel reactor protection system is that it can be tested completely during full-power operation. This capability for a thorough testing program, which contributes significantly to increasing reliability, is not possible on a 1-out-of-2 system. Topical Report, APED-5179,* presents a discussion of the reliability of the dual logic channel system.

During normal operation, all vital sensor and trip contacts are closed, and all sensor relays are operated energized. The control rod pilot scram valve solenoids are energized, and instrument air pressure is applied to all scram valves. When a trip point is reached in any of the monitored parameters, a contact opens, de-energizing a relay which controls a contact

*APED-5179, I. M. Jacobs, "Reactor Protection System, A Reliability Analysis" General Electric Co., June, 1966.

in one of the two subchannels. The opening of a subchannel contact de-energizes a scram relay which opens a contact in the power supply to the pilot scram valve solenoids supplied by its logic channel. To this point only one half the events required to produce a reactor scram have occurred. Unless the pilot scram valve solenoids supplied by the other logic channel are de-energized, instrument air pressure will continue to act on the scram valves and operation can continue. Once a single channel trip is initiated, contacts in that scram relay circuit open and keep that circuit de-energized until the initiating parameter has returned within operating limits and the reset switch is actuated manually. Reset of that circuit is possible if all parameters in that circuit are within operating limits. Once a full scram is initiated (i.e., one in channel A and one in channel B) reset is possible for each channel that has returned to operating limits. The electrical logic indicates that if a scram conditon occurs simultaneously in both channels A and B, scram valve sequences are initiated to drive the control rods into the core. Should one of the scram channels then become clear (i.e., within operating limits) and if at this time the reset switch is manually actuated, the scram condition is removed from all four rod groups. Rod motion at this time is a function of the time after scram signal, control rod dynamics, rod positon, prescribed procedures and operator action. If the scram is initiated by the mode switch (i.e., from "RUN" to "START" to "REFUEL" to "SHUTDOWN") the scram cannot be reset until the time delay in the "Shutdown Scram Reset Interlock" has timed out. This time delay is nominally sufficient to allow full insertion of the control rods at which time reset of the scram will have no direct effect on the control rods. A failure of any one reactor trip system input or component will produce a trip in just one subchannel of one logic channel, a situation insufficient to produce a reactor scram. This resistance to spurious scrams contributes to plant safety, since unnecessary cycling of the reactor through its operating modes would increase the probability of error or actual failure.

Since each control rod is scrammed as an independent unit, the failure of any one rod to scram does not affect the ability of the other rods to scram.

The following parameters enter the Reactor Trip System chain:

1. High neutron flux. To prevent fuel damage resulting from bulk power increases, high neutron flux will initiate a scram. The nuclear instrumentation provides high neutron flux trip signals. Four IRM channels and four APRM channels are connected to each of the dual logic channels. Whether the IRM or APRM trip inputs initiate a scram is determined by the mode switch position.

2. High reactor pressure. An increase in reactor vessel pressure threatens the integrity of the reactor vessel (an important barrier to the uncontrolled release of fission products). The high pressure scram terminates the pressure rise before reactor vessel damage occurs. The referenced drawings do not indicate a recirculation pump trip to assist the termination of the pressure rise. The referenced Commonmwealth Edison letter (Ref. 10) indicates an autumn 1980 refueling outage schedule for incorporating a recirculating pump trip modification to the Dresden II plant.

3. High primary containment system pressure. Abnormal pressure could indicate a rupture of, or excessive leakage from, the reactor coolant system into the drywell structure.

4. Low reactor water level. This scram signal assures that the reactor will not be operated without sufficient water above the reactor core.

5. Control rod system scram discharge volume high level. This scram signal assures that the reactor will be operated with sufficient free volume in the scram discharge system, if properly vented, to receive the control rod drives discharge upon scram.

6. Main condenser low vacuum. This scram signal anticipates loss of the main heat sink which would result in a reactor vessel pressure rise as the condenser is isolated to protect it from overpressure. The effects of increased reactor pressure rise are discussed in parameter 2.

7. Main steam line high radiation. The radiation monitors at each of the main steam lines near the primary containment system inboard isolation valves will scram the reactor on a high radiation signal. High steam line radiation is indicative of fuel failures; a scram is necessary to prevent further fuel damage.

8. Loss of a-c power to the protection system. All electronic trips, logic relays, and scram solenoid valves will operate due to loss of power, as the Reactor Protection System M-G sets coast down and trip on loss of a-c power.

9. Partial closure of main steam line isolation valves. This scram signal assures that the reactor will not be operated without its main heat sink, since the resulting reactor vessel pressure increase could cause a fuel-damaging power transient as described in parameter 2. There are four main steam lines with two valves per line. The logic is arranged such that the partial closure of either the inboard or the outboard valve in any three steam lines (i.e., if any combination of three of the steam lines is being closed by a main steam line isolation valve) will initiate a scram. This scram is bypassed when the reactor pressure is below 600 psig.

10. Generator load rejection. A loss of generator load will cause the turbine-generator to speed up. The turbine speed governor will react by closing the turbine admission valves. The reduction of steam flow will cause the reactor vessel pressure to rise, and the initial pressure regulator will open the turbine bypass valves in an attempt to maintain reactor pressure constant. If the load reduction is sudden and of a greater magnitude than bypass valve capacity, the reactor pressure will rise, resulting in the condition described in parameter 2. To prevent fuel damage and the lifting of reactor safety valves, a sudden rejection of generator load will cause a scram. According to the FSAR, this condition is sensed by comparing turbine first stage shell pressure to generator electrical output. A high first stage shell pressure coincident with low generator electrical output will cause a scram. The referenced schematic drawings indicate that this scram is implemented by a pressure switch indicating loss (below 900 psig) of oil pressure at the hydraulic inlet of fast acting control values or by a position switch indicating the fast closure solenoid valves controlling fast closure of the turbine control valves are energized and move. This scram is bypassed when the first stage turbine presssure corresponds to less than 45% rated steam flow.

11. Turbine stop valve closure. In order to protect the turbine, generator, output transformer, and main condenser, the four turbine stop valves are automatically closed upon certain conditions described in the FSAR for the turbine control system. The sudden closure of the turbine stop valves reduces the steam flow from the reactor and causes the reactor vessel pressure to rise. The initial pressure regulator responds to the pressure rise by opening the turbine bypass valves unless opening the bypass valves would overpressurize the condenser. If the required reduction in reactor steam flow is of greater magnitude than can be compensated by bypass valve capacity, or if the bypass valves are not allowed to open, the

reactor vessel pressure rise causes a positive reactivity insertion which would lead to fuel damage. In order to prevent fuel damage resulting from a reactor pressure rise resulting from turbine stop valve closure, the four turbine stop valves have valve stem limit switches which enter the reactor trip system logic channels and trip when the valves start to close. The logic is arranged so that the partial closure of any three of the four stop valves will initiate a reactor scram. This scram is bypassed when the first stage turbine pressure corresponds to less than 45% rated steam flow.

12. Manual. A separate scram push button is provided for each logic channel. To initiate a reactor scram, the pushbuttons for both logic channels must be pushed. The reactor is also manually scrammed when the reactor mode selector switch is moved to the "Shutdown"position, this places all the logic subchannels in scram.

There are three groups of entries to each scram channel in respect to functional testing.

1. On-off sensors that provide a scram trip function.

2. Analog devices coupled with bistable trips that provide a scram function.

3. Devices which only serve a useful function during some restricted mode of operation, such as startup or shutdown, or for which the only practical test is one that can be performed at shutdown.

The functional testing (i.e., injection of a simulated signal into the instrument primary sensor to verify proper instrument responses and trip operation) is carried out on a periodic basis as noted for each subchannel trip parameter. Each group of entries to the scram channels is covered with surveillance intervals, response time testing and bypassing noted where appropriate in Table 5-1.1. (The Plant Technical Specifications for these parameters indicate that the response times of the individual trip functions shall not exceed 0.1 second.)

TABLE 5.1.1*

SCRAM INSTRUMENTATION FUNCTIONAL TESTS
MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTR. AND CONTROL CIRCUITS

| Instrument Channel | Group (3) | Functional Test | Minimum Frequency (4) |
|---|---|---|---|
| Mode Switch in Shutdown | A | Place Mode Switch in Shutdown | Each Refueling Outage |
| Manual Scram | A | Trip Channel and Alarm | Every 3 Months |
| IRM | | | |
| High Flux | C | Trip Channel and Alarm (5) | Before Each Startup (6) |
| Inoperative | C | Trip Channel and Alarm | Before Each Startup (6) |
| APRM | | | |
| High Flux | B | Trip Output Relays (5) | Once Each Week |
| Inoperative | B | Trip Output Relays | Once Each Week |
| Downscale | B | Trip Output Relays (5) | Once Each Week |
| High Flux (15% scram) | B | Trip Output Relays | Before Each Startup |
| High Reactor Pressure | A | Trip Channel and Alarm | (1) |
| High Drywell Pressure | A | Trip Channel and Alarm | (1) |
| Reactor Low Water Level (2) | A | Trip Channel and Alarm | (1) |
| High Water Level in Scram Discharge Tank | A | Trip Channel and Alarm | Every 3 Months |
| Turbine Condenser Low Vacuum | A | Trip Channel and Alarm | (1) |
| Main Steamline Isolation Radiation (2) | B | Trip Channel and Alarm (5) | Once Each Week |
| Main Steamline Isolation Valve Closure | A | Trip Channel and Alarm | (1) |
| Generator Load Rejection | A | Trip Channel and Alarm | (1) |
| Turbine Stop Valve Closure | A | Trip Channel and Alarm | (1) |
| Turbine Control-Loss of Control Oil Pressure | A | Trip Channel and Alarm | (1) |

*Extracted from Table 4.1.1 Dresden Station Unit II, Plant Technical Specifications, Change #16, November 1971.

TABLE 5.1.1 (Continued)

NOTES:

1. Once per month until sufficient exposure hours have been accumulated and interpretation of failure rate curves to give an interval of not less than one month nor more than three months. The compilation of instrument failure rate data may include data obtained from other Boiling Water Reactors for which the same design instrument operates in an environment similar to that of Dresden Unit 2.

2. An instrument check shall be performed on low reactor water level once per day and on high steamline radiation once per shift.

3. The three groups are:

   A. The sensors that make up group (A) are specifically selected from among the whole family of industrial on-off sensors that have earned an excellent reputation for reliable operation.

   B. Group (B) devices utilize an analog sensor followed by an amplifier and a bi-stable trip circuit. The sensor and amplifier are active components and a failure is almost always accompanied by an alarm and an indication of the source of trouble. The bi-stable trip circuit which is a part of the Group (B) devices can sustain unsafe failures which are revealed only on test. Therefore, it is necessary to test them periodically.

   C. Group (C) devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. The only test that is meaningful is the one performed just prior to shutdown or startup, i.e., the tests that are performed just prior to use of the instrument.

4. Functional tests are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.

5. This instrumentation is exempted from the Instrument Functional Test Definition (Section 1.F of Dresden II Plant Technical Specifications). This Instrument Functional Test will consist of injecting a simulated electrical signal into the measurement channels.

6. If reactor start-ups occur more frequently than once per week, the functional test need not be performed; i.e., the maximum functional test frequency shall be once per week.

All control rods are tested for scram times at each refueling outage. Fifty percent of the control rods will be checked every 16 weeks to verify the performance so that every 32 weeks all of the control rods have been tested.

All reactor vessel instrumentation inputs to the reactor protection system operate on a pressure or differential pressure signal. These devices are piped so that they may be individually actuated with a known pressure (or differential pressure) signal during functional testing to initiate a protection system single logic channel trip. Other on-off devices are tested similarly with basic signals.

Analog devices, notably the the flux monitoring channels, are tested in two phases. First, the device must show reasonable agreement with other similar devices and must respond normally to power level changes and control rod movements. Second, a dummy electrical signal may be introduced which uses some or all of the amplifier already tested. This dummy signal is adjusted until the set point limit is exceeded to initiate a single logic subchannel trip. These instrument subchannels are exempt from the Instrument Functional Test definition. The Instrument Functional Test for these subchannels will consist of injecting a simulated electrical signal into the measurement subchannels and is performed on a one-week cycle.

Other than the mode selector switch, the Intermediate Range Monitor (IRM) trip is only active during restricted modes of operation. The IRM is required in the "Refuel" and "Start/Hot Standby" modes only and the only meaningful tests that are performed are those just prior to use. The IRM system provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges. This instrumentation is exempted from the Instrument Functional Test definition. The Instrument Functional Test used consists of injecting a simulated electrical signal into the measurement subchannels and is performed before each startup or a maximum of once per week.

## 4.2 SYSTEM DESCRIPTION (ESF/CONTAINMENT SPRAY SYSTEM)

The functional requirements and performance characteristics of the engineered safety features (ESF) serve no function which is necessary for normal station operation. They are included in the plant for the sole purpose of reducing the consequences of postulated accidents. This part of the report is concerned with the containment spray system portion of the ESF and the licensing criteria will be applied only to the containment spray system here.

The major equipment of the entire low pressure coolant injection (LPCI)/ containment cooling subsystem consists of two heat exchangers, four containment cooling service water pumps, four main system pumps, two drywell spray headers, and a suppression chamber spray header. Full capacity flow for the LPCI subsystem (i.e., 14,500 gpm against a system head of 20 psig) is provided by operating three of the four main system pumps. The containment spray subsystem and the low pressure coolant injection (LPCI) subsystem share the same pumps and heat exchangers and the functions performed are determined by valve sequencing. The function of the containment spray is to reduce pressure in the primary containment caused by postulated accidents. During LPCI subsystem operation, water is taken from the suppression pool and is pumped into the core region of the reactor vessel via one of the two recirculation loops. (There is also a connection on the condensate storage tank to make condensate available for use in functional testing of the system.)

The initiating logic to start the LPCI pumps is a form of the one-of-two-twice logic basically requiring the LPCI pump and valve selector switches to be in "AUTO" and either low-low reactor water level and reactor low pressure or 2 or greater psi high drywell pressure to be present. Since the LPCI flow passes through heat exchangers, heat may be rejected from the containment by starting the containment cooling service water pumps to cool the heat exchangers when sufficient electrical power is available. The valving to containment spray from the LPCI pumps is accomplished at operator's discretion. Interlocks (low water level inside shroud) are provided to prevent LPCI flow from being diverted to the containment spray system unless the core is flooded. A key lock switch permits these interlocks to be overridden if containment pressure is high (greater than 1 psig).

The LPCI/containment cooling system is designed so that each component of the system can be tested and inspected periodically to demonstrate availability of the system. The Plant Technical Specifications indicate that a logic system functional test and simulated automatic actuation test of the LPCI portion of the system is completed at each refueling outage. Testing of the operation of the valves required for the various modes of operation of the system will be performed at this time. A design flow functional test of the LPCI and containment cooling water pumps will be performed once each quarter during normal plant operation by taking suction from the suppression pool and discharging through the test lines back to the suppression pool. The discharge valves to the reactor recirculation loops remain closed during this test and reactor operation is undisturbed. An operational test of these discharge valves will be performed by shutting the downstream valve after it has been satisfactorily tested and then operating the discharge valve. The discharge valves to the containment spray headers are checked in a similar manner by operating the upstream and downstream valves individually. All these valves can be actuated from the control room using remote manual switches. Control system design provides automatic return from test to operating mode if LPCI initiation is required during testing. The surveillance interval for the instrumentation for the ECCS is noted in Table 5-2.1.

TABLE 5.2.1*

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CONTAINMENT COOLING SYSTEMS INSTRUMENTATION

| Instrument Channel | Instrument Functional Test (2) | Calibration (2) | Instrument Check (2) |
|---|---|---|---|
| ECCS INSTRUMENTATION | | | |
| 1. Reactor Low-Low Water Level | (1) | Once/3 Months | Once/Day |
| 2. Drywell High Pressure | (1) | Once/3 Months | None |
| 3. Reactor Low Pressure | (1) | Once/3 Months | None |
| 4. Containment Spray Interloc | | | |
| a. 2/3 Core Height | (1) | Once/3 Months | None |
| b. Containment High Pressure | (1) | Once/3 Months | None |
| 5. Low Pressure Core Cooling Pump Discharge | (1) | Once/3 Months | None |
| 6. Undervoltage Emergency Bus | Refueling Outage | Refueling Outage | None |
| 7. Sustained High Reactor Pressure | (1) | Once 3/Months | None |

NOTES:

1. Once per month until sufficient exposure hours have been accumulated and interpretation of failure rate curves give an interval of not less than one month nor more than three months. The compilation of instrument failure rate data may include data obtained from other Boiling Water Reactors for which the same design instrument operates in an environment similar to that of Dresden Unit II.

2. Functional test calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations shall be performed during each startup or during controlled shutdowns with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per week. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

*Extracted from Table 4.2.1 Dresden Station Unit II, Plant Technical Specification, Change #16, November 1971.

# 5. EVALUATIONS AND CONCLUSIONS

## 5.1 EVALUATION AND CONCLUSIONS (RTS)

The reactor trip system electrically is the dual logic reactor protection system and as such can be tested completely during full-power operation. The Plant Technical Specifications indicate a requirement for test of each of the scram parameters on a frequency as shown in Table 5-1.1. The variables for scramming are introduced as noted in the table. The individual control rods are tested for scram operability during the operating cycle and for scram times during the refueling outage. The Plant Technical Specification for the parameters that enter the scram chain indicates that the response time of the individual trip functions should not exceed 0.1 second. Neither a procedure for measurement of, nor frequency of, observation of the response time of the trip functions was located. The response (and travel) time measurement of the scram of the control rods is performed at least at each refueling outage and the required performance is within the time used for the analytical treatment of transients.

The test conditions for the various parameters are inserted in the sensors so that scram performance can be verified. The sum of the tests indicates sufficient overlap through the activated scram of the control rods to comply with the end-to-end criterion. The reactor trip system surveillance testing interval is extracted from the Plant Technical Specification and summarized in Table 5.1.1. Not available were references to the response time measurement of the individual trip functions.

Based on the information available, it is concluded that the reactor trip system meets the current licensing criteria listed in Section 2 of this report except for instrument response time testing.

5.2 EVALUATION AND CONCLUSIONS (ESF/CONTAINMENT SPRAY SYSTEM)

The testing of all portions of the ESF/Containment Spray System is called for in the Plant Technical Specification. A logic system functional test and simulated automatic actuation test of the LPCI portion of the system is completed at each refueling outage. Also testing of the operation of the various valve sequences is performed at this time. With the one-of-two-twice logic, the instruments and parameters to automatically initiate the LPCI can be tested and calibrated and the Technical Specifications (extracts appropriate to this are in Table 5.2.1) indicate periods for this to be done. The LPCI and containment cooling water pumps are required to have a quarterly flow check. The containment cooling service water pumps supply the water from the crib house for the containment cooling heat exchangers which could then be used for heat exchange performance verification when the service water pumps are tested. The operations of the valves to direct flow for LPCI or containment spray are tested by appropriate valve sequencing and overlap testing.

Response time testing requirements for the sensors for the containment cooling were not found in the references. The switchover from LPCI is manually initiated at operator's discretion, sometime after the water level in the reactor shroud is raised above the minimum two-thirds core height interlock to assure the core is flooded. The Technical Specifications indicate the interlock is functionally tested on an interval not less than monthly or greater than three months and is calibrated on a three month cycle. It does not appear that response time testing for the instrumentation for the containment spray system would be of value based on the manual valve sequencing required to initiate system's operation.

From the information available, it is concluded that the containment spray subsystem of the ESF meets the current licensing criteria listed in Section 2 of this report.

## 6. SUMMARY

The Dresden Station Unit II nuclear power plant complies to current licensing criteria for RTS testing as defined in Section 2 of this report except for instrument response time testing.

The plant also complies to current licensing criteria for ESF/Containment Spray System testing as defined in Section 2 of this report.

## REFERENCES


1. Code of Federal Regulations, Title 10, Part 50 (10CFR50), 1979, Appendix A, (General Design Criteria).

2. U. S. Nuclear Regulatory Commission, Regulatory Guide 1.22, "Periodic Testing of the Protection System Activation 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. Commonwealth Edison Company, Dresden Station Unit II Final Safety Analysis Report.

7. Commonwealth Edison Company, Dresden Station Unit II Technical Specifications.

8. Dresden II Mechanical Drawings: M-22, February 1978; M-26-2, June 1977; M-29, September 1977; M-34, June 1977; M-35-1, February 1978.

9. Dresden II Electrical Drawings: 12E2421, March 1971; 12E2422, August 1977; 12E2423, February 1977; 12E2435, February 1977; 12E2436, September 1976; 12E2437, September 1976; 12E2438, September 1976; 12E2438A, January 1977; 12E2439, December 1976; 12E2440, December 1976; 12E2441, December 1977; 12E2441A, December 1977; 12E2464, September 1976; 12E2465, October 1976; 12E2466, September 1976; 12E2467, January 1977; 12E2468, September 1976.

10. Commonweath Edison letter (Cordell Reed) to U.S. Nuclear Regulatory Commission (Harold Denton), March 29, 1979.

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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