PLG-0575

METHODOLOGY FOR DEVELOPING RISK-BASED SURVEILLANCE PROGRAMS FOR SAFETY-RELATED EQUIPMENT AT SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 AND 3

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by

Andrew A. Dykes

Edward L. Quinn

Prepared for

SOUTHERN CALIFORNIA EDISON COMPANY

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PLG

E IGINEERS ▪ APPLIED SCIENTISTS ▪ MANAGEMENT CONSULTANTS

Newport Beach ▪ CA

Washington ▪ DC

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EXECUTIVE SUMMARY

In recent years, the U.S. Nuclear Regulatory Commission has encouraged licensees to develop and request approval of test and surveillance practices that are adequately supported on a technical basis and that minimize risk to the public. This report presents the application of a pilot program to establish risk-based justification for the content and frequency of surveillance tests at the San Onofre Nuclear Generating Station (SONGS).

The system chosen for this demonstration was the excore nuclear instrument safety channel drawer, which provides voltage indication of neutron flux to the plant protection system and the core protection calculator for reactor trip functions. Specifically, risk-based methods were used to examine the 31-day surveillance test of this system (SO23-II-5.5 through 5.8) in relation to the safety functions of the channel, its failure history, other tests that reveal information about the channel, and the technical specification requirements.

The risk-based evaluation has revealed opportunities to both reduce the content of the 31-day test and extend its test interval to quarterly. The proposed scope reductions and procedure modifications will enable the test to be accomplished without opening the safety channel drawer. This eliminates a major cause of system failures. The risk-based evaluation of surveillance intervals indicates that a quarterly test interval can be achieved without significantly increasing the overall unavailablity of the system to produce its safety function trips.

APPROACH

A risk-based evaluation of surveillance tests can be approached at many levels. The ultimate risk measure is the health effects on the public. Because core damage releases radioactive material from the fuel that could result in health effects if not contained, the severity and frequency of core damage are also used as measures of public risk. In the absence of a probabilistic risk assessment for SONGS, these risk measures cannot be used directly. For this pilot study, the unavailability of the excore nuclear instrumentation safety channel drawer to produce a proper output voltage when required for reactor trip was chosen as the risk measure. Given that all other factors remain the same, an increase in this unavailability will increase the core melt frequency. The criterion for the evaluation is that recommendations should result in no significant increase in system unavailablity to perform its safety function.

The analysis was accomplished in two stages. First, the effectiveness of the test for verifying that the safety channel could accomplish its safety function was evaluated. The functions of various component parts of the system were identified. Then, the means by which these functions are verified were identified. The operating history of the safety channel drawers was reviewed to identify the types of failures that have occurred and how they were revealed. The content of the 31-day test was then correlated with this information, and test effectiveness was evaluated for verification of safety functions and duplication with other surveillance tests and operational checks.

Following the qualitative evaluation of test effectiveness, a quantitative evaluation of the surveillance test interval was accomplished by estimating, from both generic and plant-specific data, the time-dependent and test-related failure parameters. The SOCRATES computer code was used to conduct time-dependent unavailability analyses to determine the sensitivity of average system unavailability to surveillance test interval; channel bypass time; and between test time-related "standby" failures.

RESULTS AND RECOMMENDATIONS

The results of the risk-based evaluation indicate that there is considerable opportunity for reducing the content and extending the frequency of the 31-day excore safety channel drawer surveillance test, while maintaining the unavailability of the system at or below its current level. The evaluation has generated recommendations in five areas:

- Reduction of Test Content. Test scope can be reduced by eliminating the following sections of the test, which were found to have minimal impact on the ability of the safety channel to accomplish its safety function.

  - Power Supply Voltage Verification. A support function whose acceptability is evidenced by proper channel output voltages. There have been no failures to trip as a result of out-of-specification power supply voltages. Catastrophic failures will be annunciated in the control room. This eliminates one of the sections of the test that requires opening the safety channel drawer.

  - Logarithmic Channel Functional Test. Eliminates duplication with the 31-day plant protection system test, which satisfies all of the requirements of a channel functional test. The calibration steps are required only on an 18-month interval.

  - 10^{-4}% and 55% Bistable Setpoints Tests. Both activate trip functions, but do not generate the trips. The exact power level is not critical to their safety functions, and the actual activation is annunciated in the control room. Setpoint verification requires opening the safety drawer, which is a major cause of failures in the system.

  - Linear Channel Functional Test. Eliminates duplication with the 31-day plant protection system test, which satisfies all of the requirements of a channel functional test.

- Use of Test Circuits Designed into the System. Portions of the test procedure can achieve verification of system calibration and operability by use of the test controls provided on the front panel, thus eliminating the need to open the safety channel drawer.

- Linear Channel Calibration. The equivalence of the linear calibrate circuit to a known signal, as defined by the technical specifications, was established. This permits using the linear calibrate switch provided on the front panel to satisfy the monthly calibration requirement.

- Rate channel alarm functional test. A calibration is not required on a monthly basis, and a functional test can be accomplished using the rate trip test potentiometer on the front panel.

- High Logarithmic Power Test Requirement prior to Startup. Only a channel functional test is required. This can be satisfied using the log trip test potentiometer on the front panel. The simplified steps can be made part of the operations startup procedure.

- Consolidation of Monthly Requirements into the PPS 31-Day Test. Implementation of the recommendations contained in this report will result in a much smaller procedure. The administrative burden of test setup, coordination, review, and record keeping could be eliminated by consolidating the remaining steps into the 31-day PPS test, which already accomplishes the channel functional tests. This has the disadvantage of making the purpose of the PPS test broader than originally intended.

- Extension of Surveillance Test Interval. The results of the quantitative evaluation of using the best estimate values of the failure of the system parameters are given in Table S-1. The relatively high values for system unavailability are due to the comprehensive treatment of potential common cause failures. These absolute values do not impact the results, however, since the change of unavailability with test interval is of primary interest to this analysis. Table S-1 indicates that the surveillance interval for the nuclear instrumentation excore safety channel can be extended to a quarterly interval with no significant increase in system unavailability for performing its safety function.

The failure data also indicate that the 7-day requirement for functional testing of the log high power trip prior to startup can be eliminated. Failures of the logarithmic power channel occur less frequently than those of the linear channels, so testing is done at the interval determined to be acceptable for the linear channels. However, given that the high log power trip will be one of the primary safety trips during startup, including the functional test of the log channel in the startup procedure may be prudent.

CONCLUSIONS

Several conclusions regarding the use of risk-based methods of evaluating surveillance tests can be made. First, the qualitative evaluation of test procedures versus safety functions provides valuable insights into system operation and the effect of technical specification requirements on risk. This points to areas of duplication and unnecessary detail that can be modified or eliminated. Second, the data evaluation provides

TABLE S-1. UNAVAILABILITY OF THREE-OUT-OF-FOUR EXCORE NUCLEAR INSTRUMENT SAFETY CHANNELS TO PROVIDE ACCURATE VOLTAGE OUTPUT OF NEUTRON FLUX TO THE PLANT PROTECTION SYSTEM AND CORE PROTECTION CALCULATOR (Mean Value Failure Parameters; Test Bypass Time of 2 Hours)

Test Interval (months)	System Unavailability per Demand	
	Staggered Testing	Sequential Testing
1	6.48×10^{-5}	6.51×10^{-5}
2	5.97×10^{-5}	6.07×10^{-5}
3	5.89×10^{-5}	6.06×10^{-5}
4	5.92×10^{-5}	6.17×10^{-5}
5	6.00×10^{-5}	6.35×10^{-5}
6	6.12×10^{-5}	6.60×10^{-5}

insights into test effectiveness and input for failure parameters. These insights can be important for both the qualitative and the quantitative analysis.

1. INTRODUCTION

1.1 BACKGROUND

This report documents work accomplished by Pickard, Lowe and Garrick, Inc. (PLG), and the San Onofre Nuclear Generating Station (SONGS) on a pilot program to establish a risk-based methodology for evaluating surveillance testing at SONGS. The work was undertaken to implement the recommendations of the U. S. Nuclear Regulatory Commission's (NRC) task group to the issue of surveillance testing in technical specifications (Reference 1). The thrust of the group's recommendations was that surveillance test content and frequency should have a sound technical basis. The group further stated that surveillance test requirements should not unduly consume plant personnel time or result in undue radiation exposure to plant personnel without a commensurate safety benefit in minimizing public risk.

1.2 OBJECTIVE OF PROJECT

The objective of this report is to demonstrate the feasibility of a methodology for developing risk-based surveillance programs for safety-related equipment at the San Onofre Nuclear Generating Station. This risk-based analysis enhances the effectiveness of surveillance testing by establishing a more scrutable technical basis for the procedures.

The methodology defines the safety rationale for surveillance tests and correlates test procedure steps to this rationale. Those that duplicate other procedures or provide insignificant safety impact are recommended for elimination. Surveillance test intervals that reflect a balance between the positive and negative impacts of the tests are calculated based on the generic and plant-specific failure history of the equipment.

To demonstrate its feasibility, the methodology is applied to the excore nuclear instrumentation safety channel drawer 31-day surveillance with two goals:

1. Optimize the content of the procedure with respect to the safety functions of the system, the existing technical specifications, and other associated equipment and surveillance tests.

2. Determine the surveillance interval that minimizes the unavailability of the channel to accomplish its safety function.

1.3 ORGANIZATION OF REPORT

Section 2 summarizes the risk-based methodology applied to the excore detector safety channels. Section 3 evaluates the effectiveness of the tests from a risk point of view. It defines the safety functions and correlates the testing program to those functions. Finally, it presents recommendations for consolidating the 31-day test into other procedures that accomplish the same or similar objectives. Section 4 evaluates the testing interval of the safety channels, based on failure parameters derived from both the generic data of Reference 2 and the plant-specific data of SONGS Units 2 and 3.

2. METHODOLOGY

2.1 BACKGROUND

In 1983, the NRC established a task group to address the scope and nature of problems regarding surveillance testing in the current technical specifications. The group's work and recommendations are documented in NUREG-1024 (Reference 1). In this document, surveillance requirements were defined to be "requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within the safety limits, and that the limiting conditions for operation will be met" (Reference 1, page 1-3). The document cited concerns expressed by the Committee to Review Generic Requirements (CRGR) that too-frequent testing of reactor trip system breakers and diesel generators contributes to the wear of components and unnecessary downtime. The CRGR observed that a poorly defined safety rationale was used to support particular testing requirements for these systems. It encouraged establishing better balanced test and surveillance practices aimed at improving overall safety and equipment reliability.

The recommendations of the task group are given in Table 2-1. The essence of the first and second recommendations is that both the content and frequency of surveillance testing should be based on a technical basis that minimizes risk to the public. In Section 2.3 of Reference 1, the task group stated that both engineering judgment and insights obtained from probabilistic risk assessments can be used in arriving at these judgments. It identified the FRANTIC code as one of the more promising methodologies that could be used for risk-based evaluations.

In response to the NRC initiative, the Combustion Engineering Owners Group sponsored the application of risk-based methods to justify the extension of the surveillance intervals for the reactor protection system (RPS). The resulting report, prepared by Combustion Engineering, Inc. (Reference 2), is currently under review. Although this report accounted for failure rates of the instrumentation providing signals to the RPS, it did not include a detailed examination of the instrument tests. This report provides this examination for the excore nuclear instrumentation safety channels.

2.2 TECHNICAL APPROACH

The technical approach is risk based and focuses on two rationale for establishing a surveillance test program:

1. The overall operation and test program must verify the operability of system functions that impact the safety of the plant. Within this context, the licensee may demonstrate that the safety function is available by a variety of operational checks and tests. Establishing a correspondence between operational monitoring, channel checks, functional tests, and calibrations and these safety functions can satisfy the intent of the technical specifications, while avoiding

TABLE 2-1. RECOMMENDATIONS OF THE U.S. NUCLEAR REGULATORY COMMISSION'S TASK GROUP TO STUDY THE ISSUE OF SURVEILLANCE TESTING IN TECHNICAL SPECIFICATIONS, LISTED IN ORDER OF PRIORITY (Reference 1, page 4-1)

Recommendation 1

The testing frequencies in the technical specifications should be reviewed to ensure that they are adequately supported on a technical basis and that risk to the public is minimized.

Recommendation 2

The required surveillance tests should be reviewed to ensure that important safety equipment is not degraded as a result of testing and that such tests are conducted in a safe manner and in the appropriate plant operational mode to ensure that risk to the public is minimized.

Recommendation 3

The action statements should be reviewed to ensure that they are designed to direct the plants to a safe plant operational mode in such a way that public risk is minimized and that unnecessary transients and shutdowns are precluded.

Recommendation 4

The surveillance test requirements should be reviewed to ensure that they do not unnecessarily consume plant personnel time or result in undue radiation exposure to plant personnel without a commensurate safety benefit in terms of minimizing public risk.

Recommendation 5

The preparation and organization of the standard technical specifications should be reviewed to ensure that they are consistent with 10CFR50.36 and only contain requirements that have a sound safety basis.

the potentially negative impact that duplication of surveillance testing may have on channel availability.

2. The interval at which surveillance testing is accomplished should reflect a balance between the positive and negative impacts of the test. This involves a quantitative comparison of rate at which the test reveals undetected safety function failures relative to the contribution of the test to the unavailability of the system, either due to realignment or to test-caused failures.

2.3 DEFINITION AND LEVELS OF RISK

Risk-based analysis consists of an answer to the following three questions:

- What can go wrong?
- How likely is it that this will happen?
- If it does happen, what are the consequences?

To answer these questions, one could make a list of scenarios, expressed in triplet form:

$$\langle s_i, p_i, x_i \rangle$$

where

s_i = a scenario identification or description.

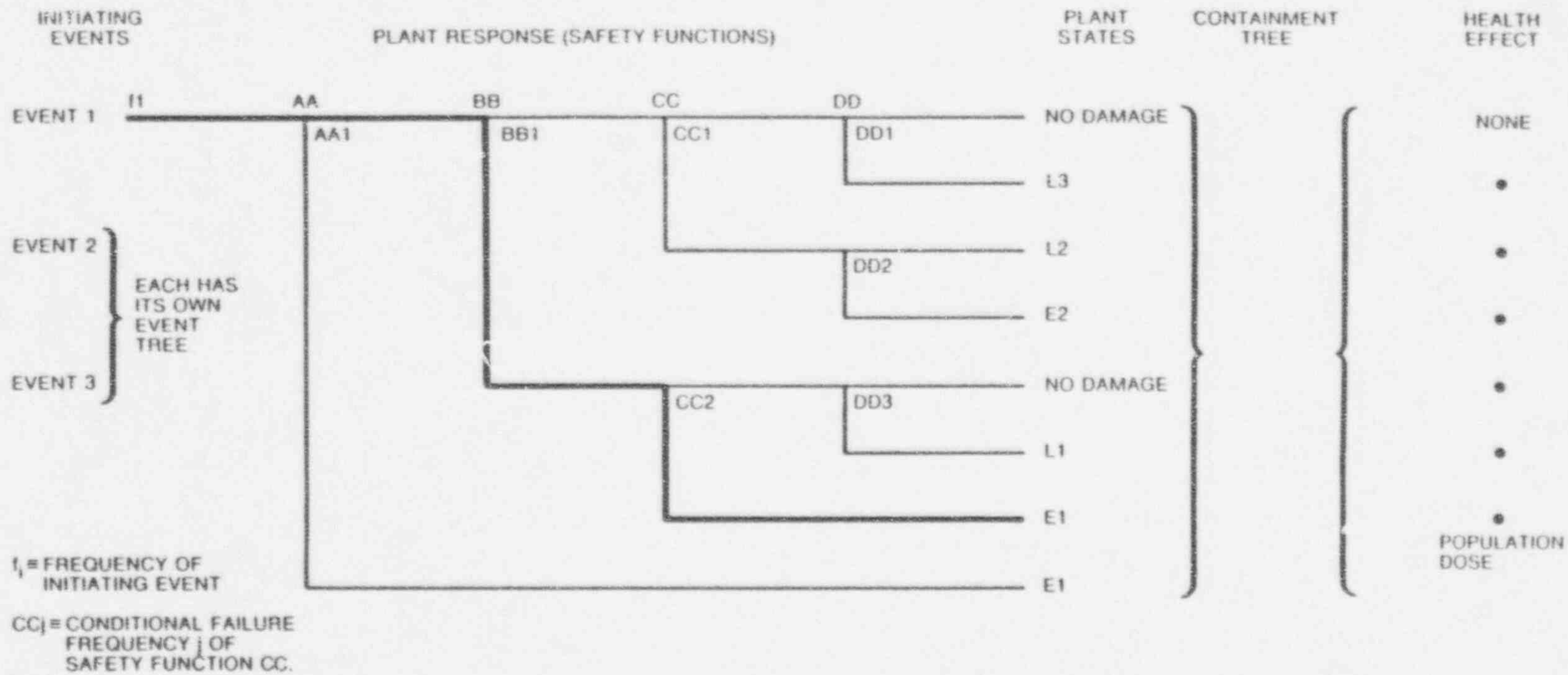
p_i = the likelihood of that scenario.

x_i = the consequence or evaluation measure of that scenario; i.e., the measure of damage.

Typically, scenarios are generated by constructing event trees that depict initiating events, the response of the engineered safety functions of the plant to those initiating events, and the end states resulting from the responses, as shown in Figure 2-1. The end states have consequences associated with them, such as health effects to the population or core damage.

Risk contributions associated with changing surveillance test intervals (STI) can be evaluated at lower levels if it can be demonstrated that the risk measure selected for evaluation has a direct relation to the overall risk described above. The two most common are the system and safety function levels. Criteria for using these measures are described in Reference 3. The following two paragraphs take much of their content from that document.

Evaluations at the safety function level address the combinations of safety systems required to perform a function that is necessary to prevent a given transient or accident from proceeding to a core melt or other undesirable consequence. The safety function is defined so that the risk impact of changing STIs can be directly tied to core melt frequency or other undesirable consequence defining the risk. The risk is an expression of the unavailability of the function, which includes



- SCENARIO S_7 IS IDENTIFIED ABOVE BY THE HEAVY LINE
- THE LIKELIHOOD OF SCENARIO S_7 IS f_1(1-AA1)(BB1)(CC2)
- THE CONSEQUENCE OF S_7 IS PLANT DAMAGE STATE E_1 (EARLY CORE MELT, EARLY CONTAINMENT FAILURE)

FIGURE 2-1. EVENT TREE RISK MODEL

all of the affected systems and their interactions. Referring to Figure 2-1, if the safety function becomes more available for accomplishing its function, the likelihood of scenarios resulting in core damage becomes smaller, leading to a decrease in risk.

System-level risk is obtained by quantifying the unavailability of a system to perform the function defined by the failure criteria of the risk analysis. Once the unavailability criteria are defined, a system unavailability model is usually easy to generate. However, when arguing the acceptability of system unavailability as the measure of risk, one needs to consider system interactions and whether more than one system is required for the successful performance of a safety function. Evaluation at the system level is generally inadequate when an STI change affects multiple systems or functions. To use system unavailability as the evaluation criteria, it must be demonstrated that the effect on system unavailability from changes in STIs can be unambiguously interpreted. This would also include not affecting initiating event frequencies or the response of other systems with which it interacts.

2.4 EFFECTIVENESS OF TESTING TO REDUCE SYSTEM UNAVAILABILITY

Surveillance testing is accomplished to demonstrate system operability and reveal system failures that have occurred but have not been revealed that would result in an unavailability to accomplish its function should an actual demand occur. To properly account for the effectiveness of the test, the source of failures and their relationship to the STI must be identified. This section first outlines the various types of failures that can occur in systems. It then summarizes how those failures might be accounted for when establishing a surveillance testing program.

2.4.1 SOURCES OF SYSTEM UNAVAILABILITY

Sources of failure to consider when evaluating STI contributions include:

- Standby Failures. Time-related between-test failures that put the system into an undetected failed state that will not be revealed until either a surveillance test is accomplished or an actual demand occurs. They are normally associated with standby equipment that remains idle until called on to operate during an emergency; hence, the name. However, these types of failures can also describe conditions under which active components or sensors must change their output in response to an emergency. If the inability to respond to the change cannot be inferred from monitored information, surveillance testing that simulates the required condition is necessary.

- Monitored Failures. Time-related between-test failures that are revealed immediately or that can be detected by the plant operators during their normal shift or daily checks. They do not require surveillance testing to be revealed.

- Demand-Related Failures. Failures that occur at specific transition times, either when the component is put into service or at the time of demand. These types of failures are normally associated with transition shocks or human errors that leave the component in a failed state. They occur independently of surveillance testing intervals and do not change as the STI changes. However, if they constitute a large fraction of observed failures, the necessity to repair demand-related failures occurring during a test is a reason for extending test intervals.

- Test-Caused Failures. Failures and degradations that require the component to be declared inoperable for repair. These failures are the result of testing and would not have occurred if the test had not been accomplished. They include human errors that require repair or otherwise increase the time during which the system is unavailable.

- Test Efficiency. Assessment of the ability of the test to reveal failure modes that will prevent successful accomplishment of the function of the system during an actual demand. This measures the ability of the test to simulate expected emergency conditions.

2.4.2 JUDGING TEST EFFECTIVENESS

An ideal test is one that

- Demonstrates the availability of the safety function.
- Does not make the system unavailable to respond to an actual demand.
- Detects failures that would not have otherwise been revealed.

In reality, tests involve a compromise of these three factors. For example, if the true alignment of the system cannot be maintained during a test, failure modes associated with that alignment may not be detected.

The failure history of the system can provide much information on the effectiveness of a test. Surveillance tests should be designed to detect conditions that cannot be revealed by monitoring or normal operational checks. If all failures are annunciated or detected by operations, the test may not be required. A preponderance of test-caused failures and demand-related failures during testing is justification for extending test intervals or seeking alternative methods of verifying operability.

Very frequently, the consolidation of tests of different systems that accomplish related functions can eliminate duplicative procedures that generate unwanted failures. This may also have the advantage of producing a better integrated verification of the safety function. The justification of an effective test should clearly state what the test is accomplishing that cannot be done by other means.

2.5 ANALYSIS FLOW

The analysis flow used in this study is given in Figure 2-2. The first few steps define the system and break down the safety functions into testable, nonredundant component functions. This forms the context under which the evaluation will be done.

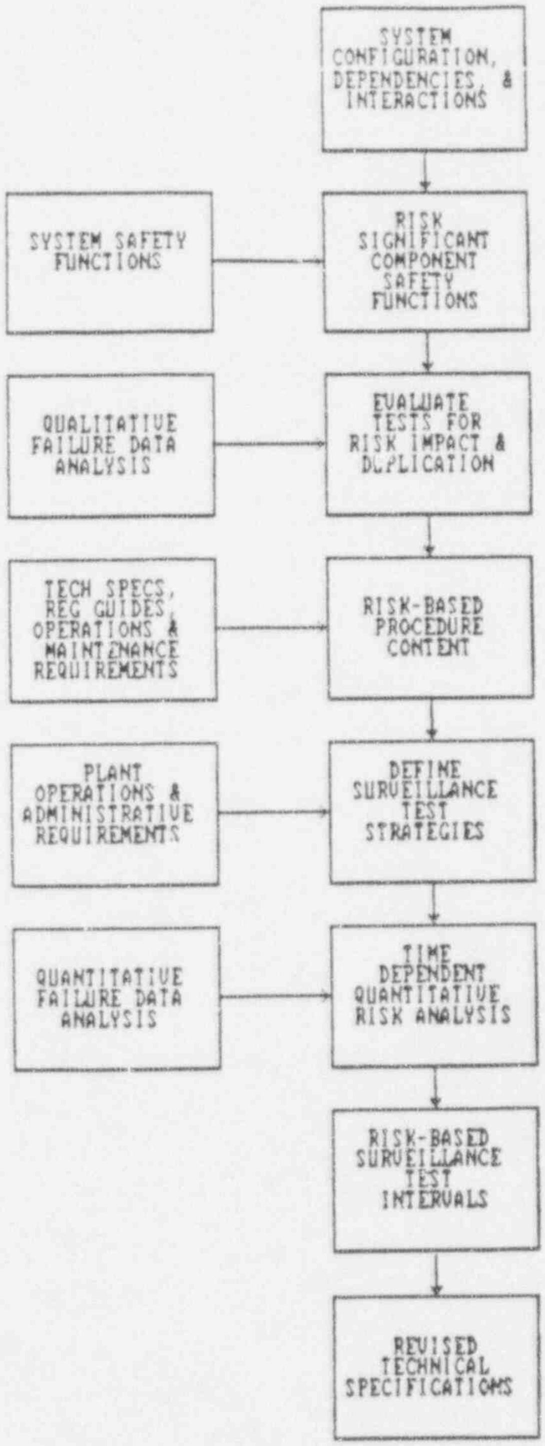


FIGURE 2-2. ANALYSIS FLOW

The next steps examine both the historical failure data and the surveillance tests that have been performed with two objectives. The first is to establish a basis for each step or section in the test procedures and validate that they are accomplishing a verification that affects the safety function of the system. The second objective is to evaluate the demonstrated effectiveness of the test to detect failures. The methods by which failures are detected are very important for analysis of test content. This evaluation can identify unnecessary and duplicative testing that can be eliminated without the necessity of a change to technical specifications.

The quantitative evaluation of surveillance test intervals requires that the failure data be broken down by the type of failure so that failure parameters suitable for use in a time-dependent unavailability analysis code, such as FRANTIC (Reference 4) or SOCRATES (Reference 5), can be used. These codes evaluate the unavailability reduction obtained from testing compared with the unavailability increase resulting from realigning the system or test-caused failures that must be repaired. It is not within the scope of this report to repeat the technical aspects of using these codes.

The quantitative analysis must account for the practical aspects of proposed testing strategies and of the administrative requirements of the plant. The application of the methodology requires engineering judgment and close coordination with the groups responsible for accomplishing the surveillance. The application to the excore nuclear instrumentation safety channel drawers provides an excellent example of the types of analysis that can be beneficial when trying to establish a rational testing program.

3. EVALUATION OF TEST EFFECTIVENESS

3.1 SYSTEM DESCRIPTION

3.1.1 GENERAL

The excore safety channels are adequately described in the applicable sections of the Southern California Edison Company (SCE) system description, SD-SO23-470, Revision 0 (Reference 6), entitled "Excore Nuclear Instrumentation System." The main functions of the safety channels are to

- Provide an assumed 0V to 10V output signal corresponding to the neutron flux power to the plant protection system (PPS) for the high linear power trip (110%) and pretrip and the high log power trip (0.83%) and pretrip

- Provide three individual subchannel 0V to 10V output signals corresponding to the neutron flux output to the core protection calculator for use in the low departure from nuclear boiling ratio (DNBR) trip and the high local power density (LPD) trip algorithms.

- Provide four channels of reactor power indication for the main control room over a range from 10^{-8} to 200% (logarithmic) and from 0% to 200% (linear).

- Provide a signal to activate the loss-of-load reactor trip circuit at 55% power.

The excore safety channel comprises two subsystems that are built into the same drawer and that share the same power supplies and detectors. These subsystems are

- Linear Power. The linear portion of the safety channel uses three vertically stacked fission chambers with no preamplification. The DC milliampere output from the detectors is converted to a 0V-to-10V output signal inside the drawer by an I/V (current-to-voltage) converter and then is summed and averaged to provide an overall linear power level signal. The average voltage output is fed to the PPS for the high linear power trip and to the control room recorders for power level indication. In addition, this signal provides input to the core vibration monitor and to the 55% loss of load bistable.

  The three individual detector output voltages are also fed to the core protection calculators for determination of the axial shape index and the calibrated excore power, which are used in the DNBR and local power density algorithms.

- Logarithmic Power. The logarithmic portion of the safety channel uses only the middle detector output through a preamplifier to the safety channel drawer. The safety channel drawer converts the preamplifier output into a logarithmic power signal using logarithmic count rate and Campbelling circuitry. This output signal is fed to the PPS for the high logarithmic power trip. It is also used for main control board indication of logarithmic power and startup rate.

3.1.2 SUBSYSTEM SAFETY FUNCTIONS

Figure 3-1, taken from Reference 6, is a simplified schematic that shows the subsystems of one excore nuclear instrumentation safety channel. It also identifies which cables and devices are in the containment, in the PPS cabinets themselves, and in the control room. The excore safety channel is fully described in the General Atomic Vendor Technical Manual (Reference 7).

Table 3-1 summarizes the contributions of the subsystems to the excore nuclear instrumentation safety channel functions under various modes of operation and power levels. Following the discussion in Section 2, the safety function of the system is to provide voltage indication of neutron flux to the PPS and core protection calculator (CPC) to trip the reactor during an uncontrolled control element assembly (CEA) withdrawal, overpower transient, or other defined operational occurrence to prevent exceeding the fuel design or reactor coolant system design limits. Four power conditions are chosen as being representative of the range of conditions for which they provide a safety function. Conclusions that may be drawn from the table are summarized below.

3.1.2.1 Logarithmic Power Channel

The primary safety function of the logarithmic power channel is to ensure the integrity of the fuel cladding and reactor coolant system (RCS) boundary in the event of an unplanned criticality from the shutdown condition. If all of the CEAs are inserted, an alarm alerts the operators to the possibility of a boron dilution incident. In the event that the CEAs are withdrawn, a high logarithmic power trip will allow them to reinsert. The most likely time that this will occur is during startup operations.

The logarithmic power circuit also provides the signal for the rate of power change alarm. When the power is low, the alarm from the rate circuit may provide sufficient time for an operator to react prior to other trips. These power levels are experienced primarily during reactor startup.

At operating power levels, the logarithmic power channel provides no safety function. However, it does provide a backup indication in the control room for the linear power level.

3.1.2.2 Linear Power Channels

The linear power amplifiers and associated circuitry provide the primary safety function of the nuclear instrumentation safety channel when the reactor has more than .83% power. They provide the proper voltage to trip the reactor and prevent exceeding the fuel design limit during overpower transients and define operational occurrences during ascent to power and normal operations.

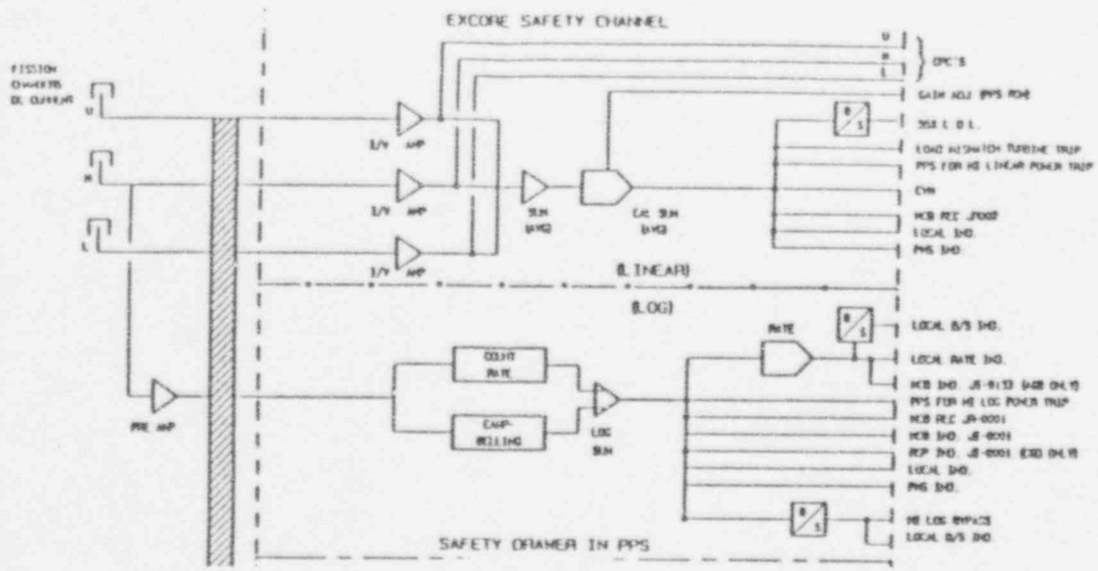


FIGURE 3-1. EXCORE NUCLEAR INSTRUMENTATION

TABLE 3-1. SIGNIFICANCE OF NUCLEAR INSTRUMENTATION SAFETY CHANNEL FUNCTIONS AT VARIOUS POWER LEVELS

Sheet 1 of 2

Safety Function \ Power Level	Reactor Shutdown (modes 3 through 6)	Reactor Startup (mode 2)	50% Power (mode 1)	100% Power (mode 1)
Log Channels				
0.83% High Log Power Trip	Trip - reinsert control rods if pulled. Alarm to operators for boron dilution events.	Trip if "overpower" during startup. Bypassed by operator during power ascent.	Bypassed.	Bypassed.
10^{-4}% Bistable Enables High Log Power Trip on Shutdown	Activates high log power trip as power declines to 10^{-4}% power.	Operator (no change) must manually bypass.	Bypasses high log power trip.	Bypasses high log power trip.
Indication to Operator of 10^{-8}% to 200% Power	Redundant indication. Startup channels primary.	Neutron flux indication to operators during power ascension.	Indication to operator, redundant with linear channels.	Indication to operator; redundant with linear channel.
Rate Indication and Alarm to Operator >2.5 decades per minute (no trip function)	Indication and alarm to operator. Impact results from operator actions.	Indication and alarm to operator. Impact results from operator action.	Indication and alarm to operator. Less time to react.	Indication to operator. Alarm has little impact due to limited reaction time.
Linear Channels (Sum)				
PPS for High Linear Power Trip at 110%	Below range. High log power trip in effect.	Becomes primary trip upon manual override of high log power trip.	Prevents exceeding a fuel design limit.	Prevents exceeding a fuel design limit.
Indication to Operator of 0% to 200%	Below range.	Redundant to log indication.	Neutron flux-based power indication to operators.	Neutron flux based power indication to operators.
Load Mismatch Turbine Trip	No function.	No function.	Input to a turbine trip.	Input to a turbine trip.
Core Vibration Monitor	No function.	No function.	Input to alarm function.	Input to alarm.

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TABLE 3-1 (continued)

Sheet 2 of 2

Safety Function \ Power Level	Reactor Shutdown (modes 3 through 6)	Reactor Startup (mode 2)	50% Power (mode 1)	100% Power (mode 1)
55% Bistable Enables Loss of Load Trip	Light "off" indicates that LOL trip bypassed.	Light "off" indicates that LOL trip bypassed.	Close to activation setpoint.	Light "on" indicates that LOL trip active.
Linear Channels (individual)				
Core protection Calculator for: • High Local Power Density Trip $>$ 21 kW/foot • Low DNBR Trip $<$ 1.31	Bypassed below 15% power.	Bypassed below 15% power.	Input for power level and axial shape calculations. Both trips required to prevent exceeding fuel design safety limits.	Input for power level and axial shape calculations. Both trips required to prevent exceeding fuel design safety limits.

Notes:

1. Trip function reference: NUREG-0741, Technical Specifications, San Onofre Nuclear Generating Station, Unit No. 2, Docket No. 50-361, Appendix A to License No. NPF-10, Amendment No. 88.

2. Other references: Technical Specification Surveillances (summarized in Appendix B).

The linear channels have only a backup safety function until the logarithmic high power trip is bypassed.

3.1.2.3 Trip Bypass Bistables

The 10^{-4}% bistable activates the logarithmic high power trip as reactor power falls below 10^{-4}% of full power. In all credible scenarios, the reactor will continue to the source range power, so the exact power level of this setpoint is not critical to its safety function. In addition, the operator can verify activation of the logarithmic high power trip on shutdown. Therefore, the exact setpoint of the 10^{-4}% bistable is not critical to the safety function of the channel as long as the activation of the function is verified.

The 55% bistable activates the loss of load trip. This circuit trips the reactor on turbine trip when the reactor power exceeds the capacity of the steam bypass control system. Since the reactor is not expected to operate at 55% power for extended periods, the exact setting of the bistable is not critical. The bypass is verified by the operators when passing through the power level, and the status of the bypass is indicated in the control room on a continuous basis. Therefore, the exact setpoint of the 55% bistable is not critical to the safety function of the channel.

3.2 CURRENT TEST PROGRAM

The current surveillance program of the excore nuclear instrumentation safety channels is directed toward satisfying the requirements of Table 4.3-1 (Reference 8) of the technical specifications, which outlines the surveillance requirements for the reactor protection instrumentation. Appendix A is a copy of this table. Understanding the definitions of those requirements and how they are currently met will assist in identifying unnecessary testing.

3.2.1 SURVEILLANCE TEST CONTENT

Table 3-2 presents the definitions contained in the SONGS technical specifications. The channel check and channel functional test definitions are straightforward, but the channel calibration definition is subject to interpretation that can significantly affect the procedures meeting its requirements. Two interpretations are important for this evaluation:

- The first sentence of the channel calibration definition states that the excore nuclear instrumentation safety channel calibration shall verify that the output voltage from the channel responds to known values of the parameter that the channel monitors. This implies that it is not necessary to verify calibration of supporting or partial subsystems if overall channel response can be verified with the necessary range and accuracy.

TABLE 3-2. SURVEILLANCE TEST DEFINITIONS
(Source: Reference 8)

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

c. Digital computer channels - the exercising of the digital computer hardware using diagnostic programs and the injection of simulated process data into the channel to verify OPERABILITY.

- The input is required to be a known value of the parameter that the channel monitors. The channel monitors neutron flux, but a known source of neutrons is impossible to obtain in an operating reactor, so the detectors are specifically excluded from the requirement. The existing procedure uses a calibrated current source to simulate detector input to the safety channel. However, there is an alternate means of producing a known current input to the channel, the linear calibrate circuit. The interpretation of the definition should account for the fact that a secondary standard is already being used.

  The use of a secondary standard to simulate a known signal is sometimes referred to as "transfer calibration." A "transfer calibration" results from an initial calibration using a National Bureau of Standards calibrated instrument or device. Then, the initial calibration is used as the standard for other applications. This concept is used for the radiation monitoring instruments and has been applied successfully in the following tests:

  - SO1-II-1.14 Unit 1 Wide Range Gas Monitor 92-Day Test
  - Palo Verde Monthly Nuclear Instrumentation Safety Drawer Calibration Test

  The equivalence of the known current source and the linear calibrate circuit as input signals to the linear amplifiers will be discussed in Section 3.3.5.

3.2.2 SURVEILLANCE REQUIREMENTS

The excore nuclear instrumentation safety channels provide input to the following functional units of the RPS:

Functional Unit	Description
2	Linear Power Level - High
3	Logarithmic Power Level - High
9	Local Power Density-High
10	DNBR-Low
18	Loss of Load

The surveillance tests that currently meet the technical specification requirements for the excore nuclear instrumentation safety channels are given in Table 3-3. A brief description and breakdown of the tests is given in Appendix B. Appendix C summarizes how these tests check the functioning of the various subsystems and components of the excore nuclear instrumentation safety channels in satisfying the surveillance requirements.

TABLE 3-3. REVIEW OF EXCORE NUCLEAR INSTRUMENTATION TECHNICAL SPECIFICATIONS VERSUS COMMITMENTS (Present)

Technical Specifications Section	Surveillance Required	Frequency	SCE Surveillance Number	Responsible Group
4.3-1 Number 2 Linear Power Level High	Channel Check	S	S023-3-3.25	OPS*
	Channel Calibration	D	S023-3-3.2	OPS
		M	S023-II-5.5 through 5.8	I&C**
		Q	S023-II-5.5 through 5.8	I&C
		R	S023-II-5.1 through 5.4	I&C
	Channel Functional Test	M	1. S023-II-1.1.1 through 1.1.4 2. S023-II-5.5 through 5.8	I&C I&C
4.3-1 Number 3 Log Power Level High	Channel Check	S	S023-3-3.25	OPS
	Channel Calibration	R	S023-II-5.1 through 5.4	I&C
	Channel Functional Test	M	1. S023-II-1.1.1 through 1.1.4 2. S023-II-5.5 through 5.8	I&C I&C
		S/U (if more than 7 days since last test)	S023-II-5.5 through 5.8	I&C

*OPS = operations.
**I&C = instrumentation and control.

Table 3-3 shows that two tests currently satisfy the requirements of the channel functional tests of both the high logarithmic power and the high linear power trips with slightly different approaches. Procedures SO23-II-5.5 through 5.8 uses both a known current source and the linear calibrate circuit to simulate 0 and 200% power level input to the channel and verifies 0V and 10V output to within the required accuracy, but it does not verify trip actuation. As one of a series of PPS checks, Procedures SO23-II-1.1.1 through 1.1.4 uses the linear trip test potentiometer as the channel input and generates a variable output voltage from the channel to verify that the high logarithmic power and linear power bistables trip at the proper voltages.

The focus of this analysis is on Procedures SO23-II-5.5 through 5.8, "Nuclear Instrumentation Safety Channel Drawer A through D Test - Linear Power Subchannel Gains - Channel Functional and Channel Calibration (31-Day interval; startup)." The consensus of plant personnel is that this test does not reflect a proper balance between the benefits obtained from revealing failures and the liabilities resulting from test-caused degradation and failures.

3.2.3 OPERATIONS AND ADMINISTRATIVE REQUIREMENTS

Table 3-4 summarizes the steps required to accomplish the monthly nuclear instrumentation safety channel test. SONGS has placed great emphasis on quality control. A permanent record of each test is maintained in the plant files, and the San Onofre Maintenance Management System (SOMMS) is used to record and maintain a detailed history of all surveillances and maintenance activities. The administrative tasks and coordination necessary to make this system the extremely useful tool that it is require a considerable amount of effort. In addition, there are administrative controls to ensure that two different technical groups do not work on the RPS at the same time.

Table 3-4 shows that the actual test is only a small portion of this effort. Given this administrative requirement, plant personnel are trying to minimize the number of different tests that must be tracked. For example, quarterly calibration requirements for the nuclear instrumentation safety channels were made part of monthly tests with the idea that it is more efficient to accomplish a few extra steps each month than to coordinate and keep administrative track of two different tests. Recommendations will recognize these practical considerations.

Bypass time is an important parameter for risk-based quantitative evaluations. Discussions with plant personnel indicate that a safety channel is normally bypassed from 1 to 4 hours for the test, with the average being about 2 hours.

3.2.4 OBSERVED FAILURES

An indication of the effectiveness of the current program of operational checks and surveillance tests may be obtained from the operational history of the excore nuclear instrumentation safety channels. SOMMS provides a detailed history of the types of failures or degraded conditions observed in the safety channel drawers and the manner by which they were detected. This database contains a record of all surveillance

TABLE 3-4. MAN-HOUR ESTIMATE FOR PERFORMANCE OF A REPRESENTATIVE NUCLEAR INSTRUMENTATION 31-DAY SURVEILLANCE (SO23-II-5.5 through 5.8)

Number	Activity	Action By	Estimated Man-Hours
1	Prepare Maintenance Order	Instrumentation and Control Planner	0.5
2	Schedule Maintenance Order	Instrumentation and Control Scheduler	0.5
3	Write Work Authorization Request (WAR)	Instrumentation and Control Scheduler	0.5
4	Approve WAR	Planning and Control (PAC)	1
5	Gather Equipment	Instrumentation and Control Technician	2
6	Prepare Surveillance Package and Transfer Data	Instrumentation and Control Technician	2
7a	Pick Up WAR (performed concurrently with 7b)	Instrumentation and Control Technician	1
7b	Issue WAR and Set Up	Operations	1
8	Perform Test (channel bypassed for 2 hours) (three men, two locations, and dual verifications required)	Instrumentation and Control Technician	6
9	Operations Support System Restoration	Operations	0.5
10a	Turn in WAR (performed concurrently with 10b)	Instrumentation and Control Technician	1

TABLE 3-4 (continued)

Sheet 2 of 2

Number	Activity	Action By	Estimated Man-Hours
10b	Close WAR and Declare "Operable," Including Performing Channel Check	Operations	2
11	Documentation Cleanup	Instrumentation and Control Technician	1.5
12	Review Surveillance	Instrumentation and Control Supervisor	1
13	Computer Entry	Instrumentation and Control Aide	0.5
14	Close WAR	PAC	0.5
	Total		21.5

$$\frac{\text{\% Man-Hours Surveillance (actual)}}{\text{\% Man-Hours Administration and Surveillance}} = \frac{6}{21.55} = 27.9\%$$

tests and all maintenance orders resulting from failures since the initial use of SOMMS in 1983.

The faults observed in the nuclear instrumentation safety channel drawers are given in Table 3-5. Each fault is assessed by the authors with respect to the type of failure mechanism involved (see Section 2.4.1) and its probability to result in a failure of the channel to produce a trip signal when a trip condition exists. This failure assessment conforms to the criteria of the quantitative model assumptions discussed in Section 4.1.2.

Of the 40 events recorded, only 11 resulted in an inoperable channel, as indicated by an assessed failure of 1.0. Of these, six were either test-caused or resulted from human error during a test. Of the remaining five, three were detected by monitoring. A fourth was found while satisfying the startup test requirement for the log circuits, but the indications available in the control room would have also revealed the failure. The fifth was revealed following a shutdown when the linear channel indicated 80% power.

It is significant that all six of the test-caused and human error failures were associated with cable connections to the back of the safety drawer. These data indicate that methods of satisfying the surveillance requirements of the technical specifications without the requirement to pull the drawer would be highly beneficial.

Only two events involving standby mechanisms, as defined in Section 2, resulted in total failure of the safety channels. First, on December 19, 1983, the channel D logarithmic power circuit on Unit 2 was found to be inoperable when the monthly test was performed to satisfy startup requirements. The assignment of one-half of the failure to monitored and one-half to standby mechanisms accounts for the fact that the failure was also revealed on the log channel indicators in the control room and could have been detected during the operator's startup procedures. The second failure (Unit 2, channel C, linear amplifier A-12 reads 80% in Mode 3, October 7, 1983) was detected by operators with control room instruments following a shutdown. The assignment of one-half of the failure to standby mechanisms conservatively accounts for the hypothesis that the channel could have been in an undetected failed state prior to shutdown and may have not responded to an upward power trend. It is important to note that a monthly test was not responsible for revealing this failure although it is assumed that, had a test been done at this time, it would have detected the failure.

Of the remaining recorded events, five involved the logarithmic calibrate circuits. Three were faults for power levels above the range of safety function applicability. The fourth was an out-of-specification reading in the test card. These faults are judged to have minimal impact on the safety function of the channel. The fifth activated a CEA prohibit, which prevents startup.

Four events involved out-of-specification power supplies, of which three were revealed by the monthly nuclear instrumentation safety channel surveillance. Two were very close to the tolerance limit, while the

TABLE 3-5. SAN ONOFRE UNITS 2 AND 3 EXCORE NUCLEAR INSTRUMENTATION SAFETY CHANNEL FAILURES THROUGH AUGUST 1988

Ch	Date	Subsystem	Deviation	Detection Method	Function Affected	Assessed Channel Failures Standby	Monitored	Test Caused	Human Error	Repair Hours	Maint Hours	MO Number
2A	06/30/85	Log Cal	Pos 5 Out Of Specification	Monthly	None	0.0					1.0	85063400000
2A	10/24/85	Log Power	Cable Induction	Operations	Log			1.0		0.5		85102293000
2A	07/18/88	Log Cal	Activate CEA Prohibited	Monthly	None			0.0			1.0	88070838000
2B	02/06/84	Ch A-12	.005V Below Specification	Monthly	Linear	0.1				12.8		84003102000
2B	06/24/84	Log Cal	Out Of Specification	Monthly	Test card	0.0					30.0	84061283000
2B	01/02/85	Linear Summer	Inoperable	Operations	Linear		1.0			7.5		85010130000
2B	12/02/85	Log Power	Broke HV Cable	Monthly	Both			1.0		4.0		85120138000
2B	02/07/86	15 VDC	At 15.219V DC	Monthly	Both	0.1				2.1		86021127000
2B	06/11/86	Linear Power	Channel 2% High	Operations	Linear		0.0				8.0	86061090000
2B	11/05/86	15 VDC	At 14.77V DC	CPC Trip	Both		0.1			6.0		86102123000
2B	11/16/87	Recorder	Reads High	Monthly	No Direct	0.0					2.0	87110546000
2B	04/21/88	10-4 Bistable	Out of Specification	Monthly	None	0.0					3.0	88041418000
2C	10/07/83	Ch A-12	Reads 80% in mode 3	Shutdown	Linear	0.5	0.5			16.4		83707354000
2C	07/01/85	Log Cal	Pos 6 Out Of Specification	Monthly	None	0.0					2.2	85070003000
2C	10/16/87	Test Circuit	AC Power Fuses Blew	Monthly	Both			1.0		24.0		87101267000
2C	03/03/88	10-4 Bistable	Out of Specification	Monthly	None	0.0					0.3	88022525000
2C	04/27/88	Rate Bistable	Out of Specification	Monthly	None	0.0					3.0	88031341000
2D	12/19/83	Log Power	Inoperable	Startup	Log	0.5	0.5			25.0		83714124000
2D	03/12/85	Log Power	Connector Failed	Monthly	Log			1.0		39.5		85031222000
2D	04/12/85	Log Power	Lead Connection	Startup	Log		1.0			72.1		85041232000
2D	07/07/85	Log Cal	Pos 6 Out Of Specification	Monthly	None	0.0					1.5	85070076000
2D	07/15/86	Log Cal	Pos 4 Setpoint	Unkown	Log				0.1			86071262000
2D	12/04/87	Linear Power	Out of Specification	Monthly	Linear	0.1					30.0	87112721000
3A	05/15/84	10-4 Bistable	Reset	NA	None						16.0	84051271000
3A	11/26/84	AC Power	Short to Ground	Monthly	Both			1.0		27.4		84112705000
3A	04/28/85	15 VDC Power	At 14.78V DC	Monthly	Both	0.1				10.4		85042727000
3A	11/14/85	Log Preamp	Reads High	CPC Test	Log	0.1				10.7		85111781000
3A	03/07/86	15 VDC Power	Neg -14.30V DC	Monthly	Both	0.1				9.5		86023262000
3A	09/04/86	Ch A-11, A-12	9.938V DC at 200%	Monthly	Linear	0.1				3.0		86090373000
3A	03/19/87	Lin Pwr Meter	Reads Low by 4%	Operations	None		0.0				9.2	87031749000
3A	11/30/87	Ch A-10, A-11	Out of Specification	Monthly	Linear	0.1				0.2		87110983000
3B	03/13/86	Linear Power	Conn P-8 Shorted	Monthly	Linear			1.0		3.3		86031299000
3B	02/12/87	15 VDC Power	Overcurrent Prot	Operations	Both		1.0			6.9		87020871000
3B	12/06/87	55% Bistable	Setpoint Out of Specification	Monthly	None	0.0					4.0	87111684000
3C	09/04/86	Linear Power	Pretrip Setpt Change	NA	None						1.0	86090327000
3C	10/11/86	Test Circuit	Fail to Energize	Monthly	None			0.0			6.0	87100115000
3C	07/06/88	Recorder	Found Turned Off	Monthly	No Direct	0.0					19.0	88062268000
3D	05/25/85	Ch A-10	10.061V DC at 200%	Monthly	Linear	0.1				1.7		85052460000
3D	09/08/86	Linear Power	Pretrip Setpt Change	NA	None						1.0	86090328000
3D	08/06/88	Ch A-10	Out of Specification	Monthly	Linear	0.1				11.0		88070831000
						Total 2.0	Total 4.1	Total 6.0	Total 0.1	Average 14.0	Average 8.1	

Ch = Channel

third was 0.7V below the nominal. There was no discrepancy in channel output observed before the test, so it is judged that these faults would have only a small likelihood of preventing the channel from producing a trip signal at the proper power level. The assignment of fractional standby failures of 0.1 to these events reflects what is believed to be a conservat' e assessment of the likelihood that they would in fact result in a fail. e of the safety function. Since surveillance tests are suspended until a discovered fault is corrected, data on the response of the channels under degraded power supply conditions are not available in the test data. However, it should be noted that these conditions were not sufficient to cause out-of-specification readings during channel checks by operators prior to the tests. The fourth power supply fault was listed as a potential cause of a spurious CPC trip. An out-of-specification power supply was found during the investigation of that event.

There were seven instances of out-of-specification individual linear channels. These are judged to slightly change the power level at which the high power trip would occur, but have only a slight likelihood of making the trip safety function occur at a power level that would increase the risk of core damage. The fractional failure assessment reflects this judgment.

Finally, 17 events were judged not to be functional failures, but required the channel to be bypassed for maintenance.

Based on the observed failure data and the manner in which they have been detected, it appears reasonable to conclude that the monthly surveillance tests have revealed relatively few potential failures. In contrast, they have been a major contributor to failures actually observed in the safety channels.

3.3 EVALUATION OF POTENTIAL MODIFICATIONS OF PROCEDURES S023-II-5.5 THROUGH 5.8 WITHIN CURRENT TECHNICAL SPECIFICATIONS

The preceding sections have discussed a variety of considerations that may be taken into account in establishing a risk-based approach to meet the current plant technical specifications. This section makes specific recommendations for the testing of the various subsystems and individual channels in the excore nuclear instrumentation safety drawer based on those discussions. Before addressing each part of the test, a few general comments will be made.

First, a majority of the failures that resulted in an inoperable channel were the result of sliding out the safety channel drawer and removing connectors to accomplish tests. As a result, there is strong justification for developing procedures that can accomplish the equivalent of the existing tests without requiring that the drawer be disturbed. The design of the drawer has provided calibration and functional test circuitry with access on the front panel, and these should be used if it can be shown that they will not increase the potential for failures.

There appears to be sufficient overlap between the excore nuclear instrumentation safety channel test and the PPS test to warrant consolidation. This will require that the steps necessary to satisfy both the overall and subchannel calibration requirements be added to the PPS test. There appears to be reasonable justification for accomplishing these checks without removing the safety drawer from its tray, so the change will not involve much additional time or many steps. However, it does slightly divert the PPS test from its primary purpose and may not be desirable from an administrative point of view.

Changes in the surveillance interval or type of surveillance that require a technical specification change will be addressed in Section 4. Specifically, the failure data support an extension of the nuclear instrumentation safety channel test interval to 90 days. At this extended interval, it may be administratively advantageous to retain a separate test for the nuclear instrumentation safety channels. However, the recommendations below for simplifying the procedure would also apply at the quarterly test interval.

As outlined in Appendix B, Procedures S023-II-5.5 through 5.8 address the functioning of all its subsystems and components in a series of test sections. These will now be addressed individually.

3.3.1 SECTION 6.2.1 - POWER SUPPLY CHECKS

3.3.1.1 Recommendation

Delete performance of this check for both the ±15V and 800V power supplies on a monthly basis.

3.3.1.2 Justification

The power supply is currently aligned on an 18-month basis. Low voltage of the 800V power supply is annunciated in the control room. Power supply voltage is a support function with no direct output to other systems. Therefore, there is no specific technical specification requirement to verify its accuracy on a 31-day basis. Although the 15V DC power supply has been found to be out of specified voltage range during monthly tests, the acceptable accuracy of the amplifiers that they power provides adequate evidence that the power supplies have not drifted significantly. If they were to drift excessively, the linear subchannel gain would not be in tolerance. Finally, checking power supply voltages requires opening the safety drawer, which increases test-caused failures. In light of the above discussion, elimination of the power supply checks from the monthly tests would be consistent with recommendation 2 of Reference 1 (see Table 2-1).

NOTE: At Palo Verde, the power supplies are checked on an 18-month interval (see Appendix D). The following information provided by Palo Verde may also be useful. An analysis program was conducted of a single drifting ±15V power supply, and the root cause was found to be a buildup of dust on the voltage adjustment potentiometer. This can be reduced significantly by "wiping" the potentiometer during the 18-month calibration. Rotating the potentiometer all the way clockwise, then counter-clockwise, successfully minimizes the buildup of dust in the contact surfaces. It is recommended that this be added to the SONGS 18-month surveillance (SO23-II-5.1 through SO23-II-5.4).

3.3.2 LOGARITHMIC POWER CIRCUIT

This check satisfies both the monthly and the startup functional test requirements. Each requirement will be addressed.

3.3.2.1 Recommendation

- Monthly. Take credit for the functional test accomplished by SO23-II-1.1.1 through 1.1.4.
- Startup. Only a functional test is required, which can be satisfied by verifying a proper response to the signal generated by the log trip test potentiometer located on the front panel. Recommend that this test be accomplished by the Operations Department during startup, or, alternately, that the startup test requirements for the I&C procedure be changed to require just the functional test using the controls provided on the front panel. Add the necessary steps to Operations Procedures SO23-5-1.3 and SO23-5-1.3.1.
- Discrepancies. If discrepancies are observed and a maintenance order is generated, accomplish the applicable portions of the 18-month surveillance, SO23-II-5.1 through 5.4, to verify operability before returning to service.

3.3.2.2 Justification

At power, the logarithmic power channel is only a backup reading in the control room. The linear channel provides the automatic trip signal.

Since functional testing requires only operability determination, including alarm and/or trip functions, the PPS 31-day test currently satisfies this requirement.

If any adjustments or calibrations are required because of an out-of-tolerance condition, the Instrumentation and Control Department should be notified. The adjustments or recalibrations will be performed by the instrumentation and control technicians.

3.3.3 HIGH LOG POWER TRIP ACTIVATION, 10^{-4}% BISTABLE CHECK

3.3.3.1 Recommendations

- Verify that the bistable activates as part of the operations shutdown procedure.
- Verify the setpoint of the bistable as part of the 18-month nuclear instrumentation calibration, but eliminate it from SO23-II-5.5 through 5.8 and SO23-II-1.1.1 through 1.1.4.

3.3.3.2 Justification

It was shown in Section 3.1.2.3 that the exact power level of the setpoint is not critical to safety. The bistable is used to activate the high logarithmic power trip on shutdown, not cause the trip itself. At the 10^{-4}% power level, the reactor will be shut down and the neutron power reflects the radioactive decay of delayed neutron precursors. Only one incident of setpoint drift and no failures of these bistables have been found in SOMMS events. Therefore, the setpoint calibration check in the 18-month test is judged to be sufficient verification. This is consistent with recommendation 5 of Reference 1.

It is important for operations to verify activation of the trip circuit as the power level passes through the setpoint range. Indications are available in the control room, and activation of trip circuits will be verified independently of setpoint verification. This verification should be included in the shutdown procedure.

3.3.4 SECTION 6.2.4 - RATE CHANNEL

3.3.4.1 Recommendation

- Monthly. Change this section to provide a functional test of the rate circuit using the rate trip test potentiometer and meter on the front panel, or add the equivalent test steps to the PPS monthly test.

- Startup. Add a requirement to verify the operation of the rate alarm to Procedures S023-5-1 3 and S023-5-..3.1, using the rate trip test control and meter indication on the front panel of the safety drawer, prior to each startup.

3.3.4.2 Justification

Since the rate circuit provides only an alarm with minimal safety impact, the precision of the current test is judged to be unnecessary. The channel functional test requirement can be adequately satisfied with circuits designed into the safety channel for this purpose.

NOTE: As discussed in Section 3.1.2.1 and Table 3-1, the effectiveness of the high power rate change alarm is primarily during startup. When the reactor is at operating power levels, there will be too little time to react to the alarm and too many other indications dominating the operators' attention for the rate alarm to have a significant impact on their mitigating actions. Therefore, functionally testing the alarm as part of a startup procedure, while eliminating this requirement from any monthly surveillance test while at power, is reasonable. The current technical specifications do not specifically mention the rate alarm. However, it has conservatively been included as an alarm associated with the high logarithmic power trip. Elimination of this check would require a reinterpretation of the technical specifications, but is consistent with recommendation 5 of Reference 1.

3.3.5 SECTIONS 6.2.5 THROUGH 6.2.9 - LINEAR CHANNEL AMPLIFIER, SUMMER, AND OUTPUT BUFFER CHECKS

3.3.5.1 Recommendation

The linear channel requires both a monthly functional test and channel calibration. The recommendations follow.

3.3.5.1.1 Channel Calibration

Eliminate this requirement from the existing nuclear instrumentation monthly test. Add the channel calibration steps to the PPS 31-day functional test. Use the "linear calibrate" potentiometer on the front of the nuclear instrumentation drawer to verify calibration for "zero" and "200%," while reading the output on both the remote operator module (ROM) (for individual amplifiers) and the PPS-installed voltmeter (for summed output).

3.3.5.1.2 Functional Test

Take credit for the PPS monthly test, Section 6.6.

3.3.5.1.3 Linear Subchannel Gain

Add the verification of the linear subchannel gains to the PPS 31-day functional test. Use the "linear calibrate" potentiometer to verify the channel calibration, while reading the output on the ROM indication in the main control room.

3.3.5.2 Justification

3.3.5.2.1 Channel Calibration

Currently, the procedure uses a calibrated current source to simulate a known value of the parameter that the channel monitors. This is consistent with the calibration requirement since the detector puts out a direct current. However, this requires opening the safety drawer and disconnecting the detector input to accomplish.

The safety channel drawer design provides a calibrate circuit that injects an equivalent current into the linear amplifier from a calibrated voltage loop. The calibrated current source is currently used to set calibrate circuits with the values used to establish the shape-annealing matrix elements in the core protection calculators during the 18-month calibration, making the calibration circuit a known source. The calibrate circuit signal is injected at the input jack, as shown in Figure 3-2. This results in an equivalent input signal to the linear amplifier that is judged to satisfy the requirements of the technical specifications.

The recommendation in Section 3.3.5.1.1 includes a complete channel calibration with the same verification points as the existing 31-day test. The ROM indication in the control room can measure the voltage being input to the CPC to within 0.005V. This is well within the 0.05V

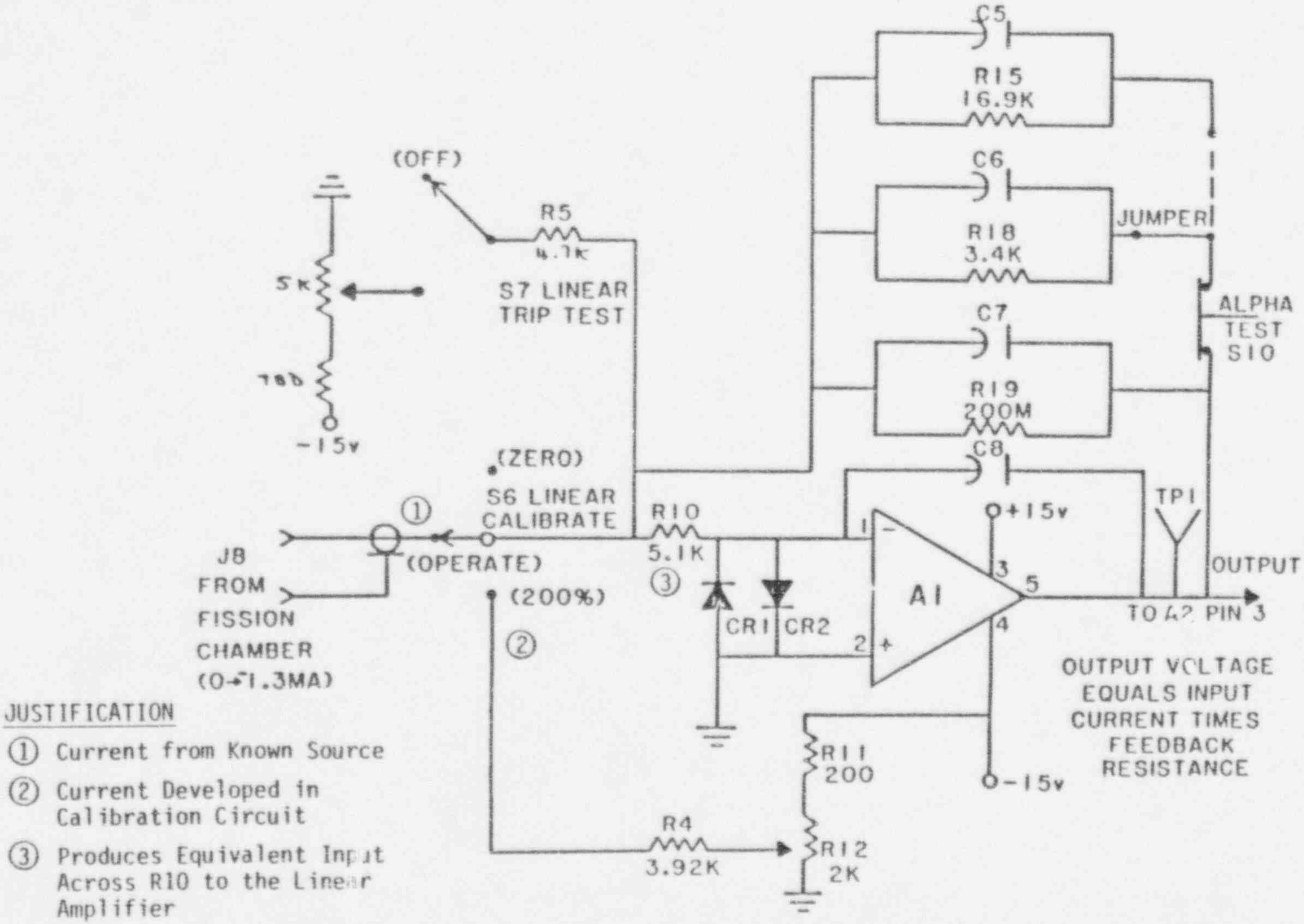


FIGURE 3-2. EQUIVALENCE OF CALIBRATION CIRCUIT WITH KNOWN CURRENT SOURCE

acceptance criterion of the channel calibration. With these supporting arguments, the concept of transfer calibration, which was discussed in Section 3.2.1, can permit calibration by using controls on the front panel without opening the safety channel drawer.

This application is further justified because

- Offsetting errors in the calibrate circuit and amplifier th· would mask amplifier problems are highly unlikely.

- The correlation of the current source to neutron flux cannot be directly established. This is recognized in the technical specifications by note 4 of Table 4.3-1, which excludes the neutron detectors from the channel calibration requirement. See Section 3.2.1 for a further discussion of this point.

If a discrepancy is found, the drawer can be opened, repaired, and checked with a calibrated current source by using applicable portions of the 18-month calibration.

3.3.5.2.2 Functional Test

Existing nuclear instrumentation and PPS surveillance overlap in meeting this requirement.

3.3.5.2.3 Linear Subchannel Gain

The intent of technical specification note 3 is to ensure that the postrefueling outage adjustment of milliampere input to voltage output correlation is still in calibration with no drift or degradation. As stated above, the 0% and 200% positions for the "linear calibrate" potentiometer are adjusted prior to startup after refueling to provide a calibrated value of milliamperes to the amplifier required for the shape-annealing matrix of the CPC.

3.3.6 SECTION 6.2.10 - 55% BISTABLE, LOSS OF LOAD TRIP ACTIVATION

3.3.6.1 Recommendation

Verify that loss of load trip circuit activates as part of startup procedures during power ascent. (Delete this section of the nuclear instrumentation surveillance.)

3.3.6.2 Justification

The check in Procedure SO23-II-5.5(-5.8) is only a verification of the power level at which the LOL trip circuit activates. It does not cause the trip. Indication of the activation of the individual channel trips are available in the control room. It was shown in Section 3.1.2.3 that the exact power level of the setpoint is not critical to safety, and operations can verify the activation of the LOL trip as the power level passes through 55%. In addition, no drift of setpoints or failure of these bistables was found in SOMMS events. Since the activation of the circuit is monitored, the calibration check in the 18-month test is judged to be a sufficient verification of the activation setpoint. This is consistent with recommendation 5 of Reference 1.

3.3.7 RECOMMENDED NUCLEAR INSTRUMENTATION EQUIPMENT MODIFICATIONS

As a result of the review of the nuclear instrumentation safety channel surveillances and the equipment failure modes, it is recommended that Raychem heat shrink sleeves be added over the field cable to the j-connector mating at the back of the safety drawers. This will strengthen the connector support and minimize connector-related failures in the future. This has been done, with good results, at Palo Verde.

4. QUANTITATIVE EVALUATION

4.1 QUANTIFICATION MODEL

4.1.1 SYSTEM MODEL

Figure 4-1 is the fault tree model of the excore nuclear instrumentation safety drawers. As stated in the system description, four physically and electrically separated channels provide voltage signals to the plant protection system and the core protection calculator. A two out of four coincidence of trip signals is required to generate a reactor trip signal. Consequently, the channels will fail to provide the required signals if three out of four channels are unavailable at the time an overpower condition requiring reactor trip occurs. This failure criterion is expressed by the top event of the fault tree in Figure 4-1.

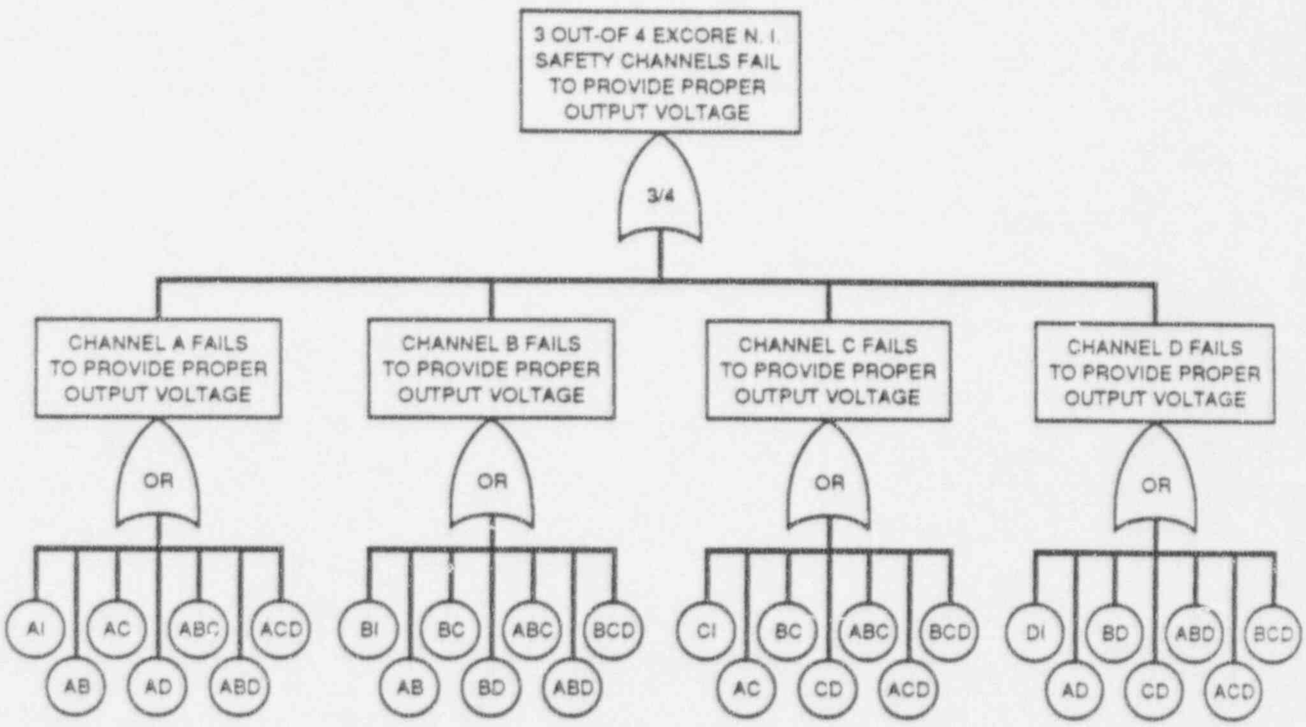
The function that this report addresses refers to the availability of the high power and high logarithmic rate of change parameters, which are just 2 of the 13 types of trip parameters listed on page 2-3 of Reference 2. The scope is explicitly limited to testing policy for the circuitry that converts the current from the detectors into voltages suitable for use by the reactor protection system. Within the block diagram given in Figure 2.1-1 of Reference 2, reproduced here as Figure 4-2, the trip parameters would be contained with RSP1 to RSP4, which represent the four independent channels of the 13-trip parameter. Within this context, the fault tree is developed to the same level of basic events as Reference 2.

Because of the limited scope of this study, the unavailabilities resulting from the fault tree are conservative since the cutsets resulting from the fault tree are not sufficient to fail the trip parameter portion of reactor protection function. For example, a high overpower transient is expected to produce an over-pressure condition as well as an increase in neutron flux. To the extent that the diverse parameters will respond to an initiating event, the cutsets for the trip function will require more simultaneous failures or dependent failures. Consequently, the use of the partial fault tree will indicate a larger magnitude change in unavailability as a result of a change in the testing policy for the NI safety drawers than will a complete model of the reactor protection system.

The assumptions in Section 4.1.2 recognize the potential for interactions between the excore instrumentation safety drawers and other systems. However, since the safety drawers are individual pieces of electronic equipment, the potential for these interactions are considered to be very small and will not impact the decision regarding the testing policy. This judgment is supported by the review of industry failure data, which found no common cause failures of the safety drawers and other systems.

4.1.2 ASSUMPTIONS

1. Failure of an individual excore nuclear instrumentation safety channel is a failure to output the proper voltage during a power



KEY
AI, BI, CI, DI – INDEPENDENT CHANNEL FAILURES
AB, AC, AD, BC, BD, CD – DOUBLE COMMON CAUSE FAILURES
ABC, ABD, ACD, BCD – TRIPLE COMMON CAUSE FAILURES

FIGURE 4-1. EXCORE NUCLEAR INSTRUMENTATION SAFETY CHANNEL FAULT TREE

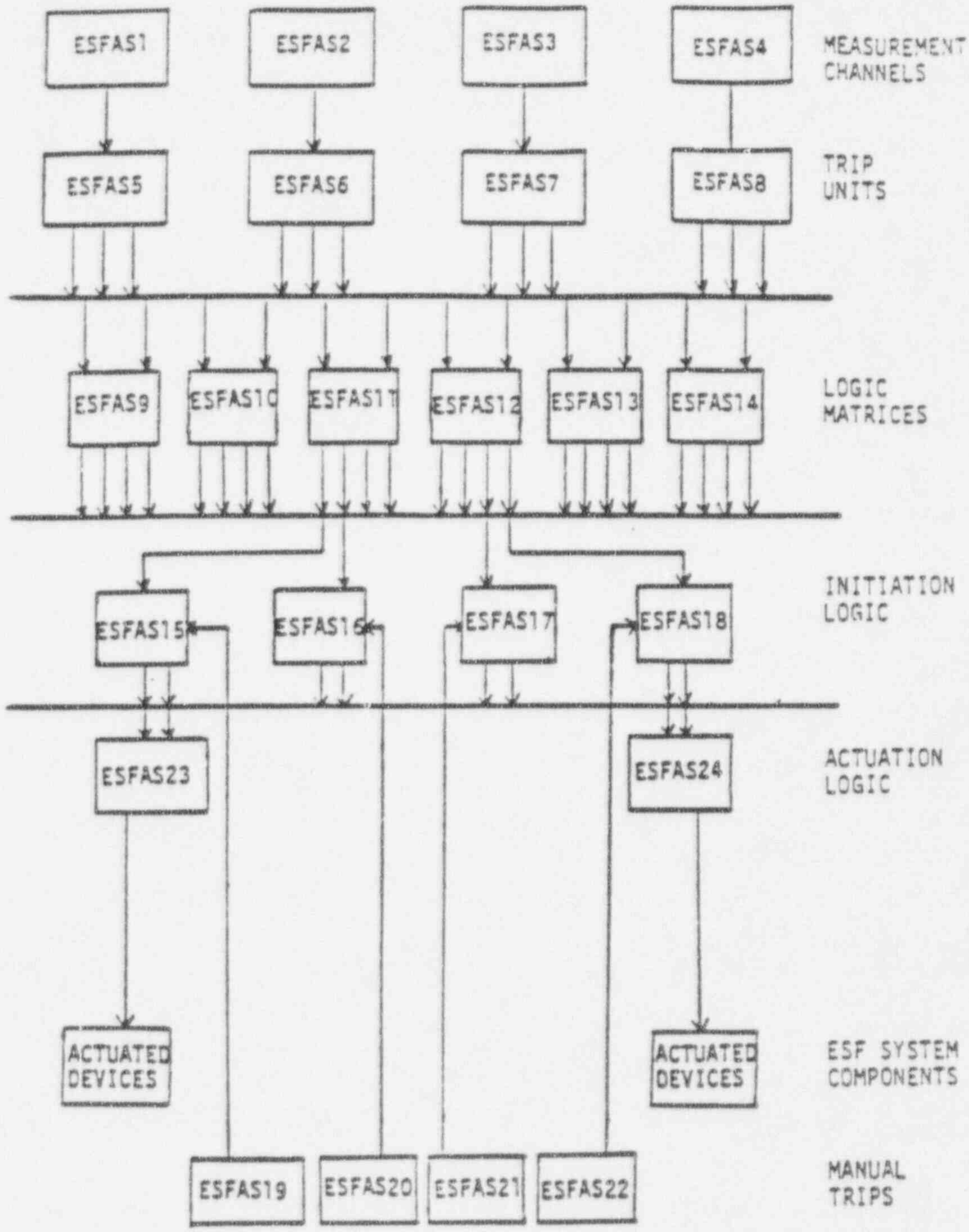


FIGURE 4-2. TYPICAL C-E DESIGNED ESFAS FUNCTIONAL BLOCK MODEL (FROM REFERENCE 2)

transient, resulting in a failure of that channel to trip before the reactor pressure safety limit or fuel design limits are exceeded. For the purposes of this analysis, the failure parameters generated in Section 4.2 result in improper voltages of the magnitude that would create these conditions.

2. Components that respond to the output voltage are considered to be outside the boundary of the system. They interact symmetrically with the channels of the system and will not affect the conclusions of this analysis.

3. During testing, a channel is bypassed and unable to produce a trip signal. This change in logic is modeled explicitly by requiring the unavailability of a channel to be 1.0 while being tested.

4. Changes in the surveillance test frequency of the excore nuclear instrument safety channel will not increase the frequency of transient initiating events. An increase in test interval may decrease the frequency of inadvertent trips due to testing, so this assumption is conservative.

5. Interactions between the excore nuclear instrument channels and other systems by any means other than providing a proper output voltage, as expressed by the system unavailability, are assumed to be negligible.

4.1.3 COMMON CAUSE FAILURES

Experience with redundant systems indicates that, despite their physical and electrical separation, the safety channels can be subject to common cause failures. Therefore, in the model the unavailability of each channel has contributions from an independent basic event and from all combinations of double and triple common cause failures that can lead to the failure of that channel.

The methods by which common cause failures can be revealed depend strongly on the mechanisms that cause them. The data analysis in Section 4.2 indicates that the likelihood of time-related standby failures being due to common cause mechanisms is much smaller than for demand-related failures. This is consistent with the difficulty involved in hypothesizing a mechanism by which a common cause time-related failure can occur in the excore safety channel drawers. It would have to create a state in two or more active detector channels that would prevent them from responding to an overpower condition but that would also remain unrevealed by the CPC or control room indicators until the next surveillance test. This type of failure would be of sufficiently unique origin that it is assumed that the potential for a common cause failure would be investigated if such a failure were found on any one channel. Therefore, the common cause failure is assumed to be revealed when any one of the channels is tested.

4.1.4 EVALUATION OF SYSTEM FAULT TREE

The cutsets that result from the evaluation of the fault tree are given in Table 4-1, which is the input echo from the SOCRATES code discussed in

TABLE 4-1. SOCRATES ECHO OF INPUT DATA FOR EXCORE NUCLEAR INSTRUMENTATION SAFETY CHANNEL UNAVAILABILITY EVALUATION (Sheet 1 of 4)

INPUT DATA

COMPONENT DEFINITION

INPUT FILE: FEB11.DAT

COMPONENT NAME	COMP. PARAM LIST INDEX	TEST INTERVAL	SCHEDULED DOWNTIME	ALLOWED OUTAGE TIME	COMPONENT UNAVAILABILITY
AI	1	730.0	2.00	14.00	
BI	1	730.0	2.00	14.00	
CI	1	730.0	2.00	14.00	
DI	1	730.0	2.00	14.00	
AB	2	730.0	2.00	14.00	
AC	2	730.0	2.00	14.00	
AD	2	730.0	2.00	14.00	
BC	2	730.0	2.00	14.00	
BD	2	730.0	2.00	14.00	
CD	2	730.0	2.00	14.00	
ABC	3	730.0	2.00	14.00	
ABD	3	730.0	2.00	14.00	
ACD	3	730.0	2.00	14.00	
BCD	3	730.0	2.00	14.00	

COMPONENT PARAMETER LISTS

INPUT FILE: FEB11.DAT

- LAMBDA- COMPONENT FAILURE RATE
- TESTQ- PROBABILITY THAT THE COMPONENT TEST CANNOT BE OVERRIDDEN
- GAMMA- PROBABILITY OF TEST-CAUSED FAILURE
- RHO- PROBABILITY OF FAILURE ON DEMAND
- K- AOT OCCURRENCE MULTIPLICATION FACTOR
- FRACQ- FRACTION OF COMPONENT FAILURES NOT

Table 4-1 (Sheet 2 of 4)

```
*        DETECTED AND NOT REPAIRED            *
* FRACG- FRACTION OF TEST-CAUSED FAILURES NOT *
*        DETECTED AND NOT REPAIRED            *
***********************************************************************************************************
*                                                                                                         *
* COMP. PARAM                                                                                             *
* LIST INDEX     LAMBDA     TESTQ      GAMMA      RHO        K          FRACQ      FRACG                   *
*     1          3.00E-06   1.00E+00   1.96E-03   8.40E-04   8.10E-01   0.00E+00   0.00E+00                *
*     2          2.30E-09   0.00E+00   1.38E-04   5.60E-05   1.00E+00   0.00E+00   0.00E+00                *
*     3          5.20E-10   0.00E+00   3.10E-05   1.26E-05   1.00E+00   0.00E+00   0.00E+00                *
*                                                                                                         *
***********************************************************************************************************


**************************************************
*               CUT SET DATA                     *
**************************************************
* INPUT FILE: FEB11.DAT                          *
***********************************************************************************************************
*                                                                                                         *
*  CUTSET   NUMBER OF                                                                                     *
*  NUMBER   COMPONENTS   COMPONENTS                                                                       *
*  ------   ----------   -----------------------------------                                              *
*                                                                                                         *
*    1          1        ABC                                                                              *
*    2          1        ABD                                                                              *
*    3          1        ACD                                                                              *
*    4          1        BCD                                                                              *
*    5          2        AB        , AC                                                                   *
*    6          2        AB        , AD                                                                   *
*    7          2        AB        , BC                                                                   *
*    8          2        AB        , BD                                                                   *
*    9          2        AB        , CD                                                                   *
*   10          2        AB        , CI                                                                   *
*   11          2        AB        , DI                                                                   *
*   12          2        AC        , AD                                                                   *
*   13          2        AC        , BC                                                                   *
*   14          2        AC        , BD                                                                   *
*   15          2        AC        , BI                                                                   *
*   16          2        AC        , CD                                                                   *
*   17          2        AC        , DI                                                                   *
*   18          2        AD        , BC                                                                   *
*   19          2        AD        , BD                                                                   *
*   20          2        AD        , BI                                                                   *
*   21          2        AD        , CD
```

Table 4-1 (Sheet 3 of 4)

```
*    22         2        AD       , CI                                      *
*    23         2        AI       , BC                                      *
*    24         2        AI       , BD                                      *
*    25         2        AI       , CD                                      *
*    26         2        BC       , BD                                      *
*    27         2        BC       , CD                                      *
*    28         2        BC       , DI                                      *
*    29         2        BD       , CD                                      *
*    30         2        BD       , CI                                      *
*    31         2        BI       , CD                                      *
*    32         3        AI       , BI       , CI                           *
*    33         3        AI       , BI       , DI                           *
*    34         3        AI       , CI       , DI                           *
*    35         3        BI       , CI       , DI                           *
*                                                                           *
*****************************************************************************
```



```
**************************************************
*        TEST ARRANGEMENT DATA                   *
**************************************************
* INPUT FILE: FEB11.DAT                          *
*****************************************************************************************************
*                                                                                                   *
* TEST ARRANGEMENT NUMBER:        4                                                                 *
*                                                                                                   *
*      TEST NAME:                 TESTA                                                             *
*      TEST INTERVAL (T):           730.0 HRS                                                       *
*      TIME OF FIRST TEST (T1):       0.0 HRS                                                       *
*      TEST TIME (C):                 2.0 HRS                                                       *
*      DOWNTIME (D):              ** WILL USE COMPONENT'S D **                                      *
*      COMPONENTS IN TEST:        AI        , AB        , AC        , AD        , ABC       , ABG       , ACD       *
*                                                                                                   *
*      TEST NAME:                 TESTB                                                             *
*      TEST INTERVAL (T):           730.0 HRS                                                       *
*      TIME OF FIRST TEST (T1):   ** NOT SPECIFIED **                                               *
*      TEST TIME (C):                 2.0 HRS                                                       *
*      DOWNTIME (D):              ** WILL USE COMPONENT'S D **                                      *
*      COMPONENTS IN TEST:        BI        , AB        , BC        , BD        , ABC       , ABD       , BCD       *
*                                                                                                   *
*      TEST NAME:                 TESTC                                                             *
*      TEST INTERVAL (T):           730.0 HRS                                                       *
*      TIME OF FIRST TEST (T1):   ** NOT SPECIFIE  **                                               *
*      TEST TIME (C):                 2.0 HRS                                                       *
*      DOWNTIME (D):              ** WILL USE COMPONENT'S D **                                      *
```

Table 4-1 (Sheet 4 of 4)

COMPONENTS IN TEST: CI , AC , BC , CD , ABC , ACD , BCD

TEST NAME: TESTD
TEST INTERVAL (T): 730.0 HRS
TIME OF FIRST TEST (T1): ** NOT SPECIFIED **
TEST TIME (C): 2.0 HRS
DOWNTIME (D): ** WILL USE COMPONENT'S D **
COMPONENTS IN TEST: DI , AD , BD , CD , ABD , ACD , BCD

CONSTRAINTS:

CONSTRAINT TYPE	TEST ARRANGEMENT AFFECTED	TESTS IN CONSTRAINT
STAGGERRED	4	TESTA , TESTB , TESTC , TESTD

TEST ARRANGEMENT TIMES AS ADJUSTED BY CONSTRAINTS:

TEST ARRANGEMENT NUMBER: 4

TEST NAME	TEST INTERVAL (T)	TIME OF 1ST TEST (T1)	TEST TIME (C)	TESTING OCCURS ONLY DURING SHUTDOWN
TESTA	730.0	0.0	2.0	NO
TESTB	730.0	182.5	2.0	NO
TESTC	730.0	365.0	2.0	NO
TESTD	730.0	547.5	2.0	NO

COMPONENT GROUPS

INPUT FILE: FEB11.DAT

COMPONENT GROUP INDEX	COMPONENTS IN GROUP
1	TESTA , TESTB , TESTC , TESTD
2	AI , BI , CI , DI

The resultant generic failure parameter was 9.5×10^{-6} per hour with 5th and 95th percentiles of 7.6×10^{-6} and 1.2×10^{-5} per hour, respectively. The remainder of this section describes how both the generic and site-specific data were interpreted to establish the parameters associated with the four failure mechanisms described above and modeled in the SOCRATES code.

The failure data used in Reference 2 to accomplish its update are given in Table 4-2. A total of 12 failures were reported as a result of 10 events. Only eight events could be confirmed by querying the electronic LER and NPRDS that are on-line at SONGS. Of these, one event (Milestone 2, 04/01/81, containment problem) did not involve a failure of the safety channel drawers. Of the remaining seven, five involved single failures and two were classified as double-component common cause.

The data used by Reference 2 appear to be incomplete. For example, failures have been recorded at SONGS 2 and 3 that were not included in the 12 failures. In addition, all the failures come from only three plants, indicating that there may be considerable plant-to-plant variability in the failure parameters. The 90% confidence interval given in Reference 2 for the Combustion Engineering plant-specific posterior distribution was based on an update with the total data from all of the Combustion Engineering plants, resulting in a range factor of only 1.2. Because of the uncertainties discussed above, it is judged to be appropriate to widen the range factor to 5, making the 5th and 95th percentiles of the distribution 4.2×10^{-6} and 2.1×10^{-5} per hour, respectively, with a mean of 1.1×10^{-5} per hour.

Although assumed to be standby failures, not all of the events in Reference 2 involve standby failure mechanisms. The descriptions of the events provided by the LER and NPRDS queries provide guidance for categorizing them into standby, demand, monitored, or test-caused for input into the SOCRATES model. Table 4-2 shows that only three of the seven events for which a reference could be obtained indicated that they could involve standby failure mechanisms. However, none of the three involved a total loss of the channel, but, more likely, caused only a slight change in the power level at which a trip signal would be generated. Consequently, the likelihood that the channel would fail to produce a trip signal due to this degraded condition is assessed at 0.1 per event. Thus, the three standby failures would produce a weighted equivalent 0.3 failures to trip. This is only about 7% of the total assessed failure likelihood for the seven events for which a reference could be obtained in Table 4-2. However, as shown in Table 4-2, as a reasonable balance between the number of events and the assessment of equivalent failures to trip, 25% of the Combustion Engineering plant-specific posterior distribution is assigned to the standby failure rate parameter.

The remaining reported events do not involve time-related failures. Estimation of test-caused failures and human error rates requires the number of tests in the data bank, and this information was not given in the report. In its absence and in recognition that monitored time-related failures are also possible, the C-E failure rate will be apportioned equally among these three failure parameters to form the

TABLE 4-2. PEER GROUP SAFETY CHANNEL FAILURE DATA

Unit	Date	Subsystem	Failure Mode	Deviation	Method Detected	Type Failure	Function Affected	Assessed Failure Likelihood	Reference	Remarks
Calvert C	06/01/81	Lin Power A	Fail to Operate	Connector Open	Startup	Test Caused	Linear	1.00	LER 317-81045	
Calvert C	12/12/79	Lin Power B	Out of Spec	Axial Flux Offset	Monthly	Standby	Linear	0.10	LER 318-79044	Channel gain adjusted. MO opened for chamber
Calvert C	08/20/81	Lin Power B	Erratic Operation	Intermittent Signal	Operations	Monitored	Linear	1.00	LER 318-81043	
Calvert C	01/14/81	Ch ? Shaping	Out of Spec	Dev from Calimetrics	Daily Cal?	Monitored	CPC	0.10	NPRDS	Suspect aging, reference not available
Calvert C	01/21/81	Ch B Shaping	Out of Spec	Dev from Calimetrics	Daily Cal?	Monitored	CPC	0.10	NPRDS	Suspect aging, reference not available
Calvert C	02/02/84	Lin Power A&B	Out of Spec	Lin Amps A&B Drift	Monthly	Standby	Linear	0.20	NPRDS 840202-1	CCF assessed 2 x 0.1 per channel
Millstone	04/25/78	Lin Power D	Failed Low	Loose Cable Plug	Startup	Human Error	Linear	1.00	LER Unknown	Reference not available
Millstone	05/22/79	Lin Power A&D	Fail to Operate	Open Connectors	Startup	Human Error	Linear	2.00	LER 336-79012	CCF of two channels, but not STI dependent
Millstone	10/23/80	Lin Power D	Fail to Operate	Cables Reversed	Startup	Human Error	Linear	1.00	LER 336-80036	
Millstone	04/01/81	Lin Power D	Out of Spec	Containment Problem	Monthly	Standby	Linear	0.10	LER 336-81016	Repair next shutdown
					Total Equivalent Channel Failures =			6.60		
					Reference Not Available =			1.20		
					Total Reference Available =			5.40		

Contribution of Standby Failures to Events for which a Reference was Available:

1. Percentage of Events

   3/7 = 43%

2. Percentage of Assessed Channel Failures To Produce Trip Signal

   .4/5.40 = 7%

prior of the plant-specific data. The data are converted by assuming one demand per 30 days of operating time, using the equation

$$q = (\lambda)(720 \text{ hour})/2$$

No failures of the logarithmic function were recorded in the generic database. For the purpose of establishing a reasonable prior to estimate this parameter, the logarithmic standby failure rate is assumed to be the same magnitude as the parameter of the linear function.

The prior distributions resulting from the above analysis are given in Table 4-3.

4.2.2 PLANT-SPECIFIC DATA

Table 4-4 is a compilation of the failure data presented in Table 3-5 to provide plant-specific data for a Bayesian update of the generic data. These data have been consolidated so that failure parameters for the logarithmic and linear power functions can be calculated individually. The results of the update are given in Table 4-5. This table shows that the failure rates for the logarithmic function are very close to those of the linear function. The overlapping 90% confidence intervals indicate that any differences are insignificant. Of the two sets, the linear power range parameters will yield the shortest test intervals since the standby failure rate is slightly higher and the test-caused failure rate is slightly lower. Therefore, it is considered to be more conservative and will be used to represent both functions in the quantification. These failure rates are good evidence that test intervals for the two power-level functions should be kept the same.

4.2.3 COMMON CAUSE PARAMETER ESTIMATION

As discussed in Section 4.1, redundant systems are subject to common cause failures that can disable two or more safety channels simultaneously. Although the excore neutron detector channels are designed to minimize this possibility, the potential for common cause failures must be considered. Consequently, it has been specifically included in the system model as shown in Figure 4-1. This section documents the development of the common cause failure parameters and resulting failure rates that are used in the quantification of the model.

The Reference 2 data development classified two of the ten events it listed as common cause. As this was considered a limited sample, a review of NPRDS was conducted for plants that contain similar detectors to provide a broader base of data for the estimate of common cause parameters. These plants included Arkansas Nuclear One Units 1 and 2, Palo Verde Units 1-3, Calvert Cliffs Units 1 and 2, Millstone Unit 2, Palisades Unit 1, Saint Lucie Units 1 and 2, Maine Yankee, and Fort Calhoun Unit 1. The data include those events that involve failures or out-of-specification conditions in either the logarithmic or linear power signals. It does not include data from tests accomplished during refueling outages. As noted in Appendix A, normal surveillance is not

TABLE 4-3. CONVERSION OF COMBUSTION ENGINEERING REPORT (REFERENCE 2) POSTERIOR DISTRIBUTION INTO PRIOR DISTRIBUTIONS FOR SURVEILLANCE TEST INTERVAL ANALYSIS (FAILURE OF AN INDIVIDUAL CHANNEL)

Description	Mean	Distribution Percentiles		
		5th	50th Medium	95th
Time-related posterial distribution	9.6×10^{-6}/hr	7.6×10^{-6}/hr	9.5×10^{-6}/hr	1.2×10^{-5}/hr
Distribution Broadened to EF = 5	1.08×10^{-5}/hr	4.2×10^{-6}/hr	9.5×10^{-6}/hr	2.1×10^{-5}/hr
<u>Assessed Prior Failure Parameters for this study (applies to both log and linear)</u>.				
Standby Failure Rate (25%)	2.7×10^{-6}/hr	1.1×10^{-6}/hr	2.4×10^{-6}/hr	5.3×10^{-6}/hr
Monitored Failure Rate (25%)	2.7×10^{-6}/hr	1.1×10^{-6}/hr	2.4×10^{-6}/hr	5.3×10^{-6}/hr
Demand Failure Rate due to Human Errors (25%)	9.8×10^{-4}/d	4.0×10^{-4}/d	8.6×10^{-4}/d	1.9×10^{-3}/d
Test-Caused Failure Rate (25%)	9.8×10^{-4}/d	4.0×10^{-4}/d	8.6×10^{-4}/d	1.9×10^{-3}/d

0910S010892

TABLE 4-4. CATEGORIZATION OF EXCORE SAFETY CHANNEL FAILURE EVENTS FOR SONGS UNITS 2 AND 3. SEE ALSO TABLE 3-5.

SONGS Unit	Channel	Service Hours Modes 1-5	Number of Tests	Assessed Failures								Failure Repair Time (hr)	Nonfailure Maintenance Time (hr)
				Standby		Monitored		Test-Caused		Demand			
				Log	Linear	Log	Linear	Log	Linear	Log	Linear		
2	A		90	0	0	0	0	1	0	0	0	0.5	1.0, 1.0
2	B	37,740 x 4	90	.1	.2	.1	1.1	1	1	0	0	12.8, 4, 2.1 6, 7.5	30.0, 2.0 8.0, 3.0
2	C		85	0	.5	0	.5	1	1	0	0	16.4, 24.0	2.2, 0.3, 3.0
2	D		84	.5	.1	1.5	0	1	0	0.1	0	25.0, 39.5, 72.1	1.5
3	A		68	.3	.4	0	0	1	1	0	0	10.4, 27.4 3.0 10.7, 0.2, 9.5	16.0, 9.2
3	B	34,790 x 4	68	0	0	1	1	0	1	0	0	3.3, 6.9	4.0
3	C		68	0	0	0	0	0	0	0	0	None	1.0,6.0,19.0
3	D		49	0	.2	0	0	0	0	0	0	1.7, 11.0	1.0
	Total	290,120 hrs	602	0.9	1.4	2.6	2.6	5	4	0.1	0	No. = 21 AV = 14.0 hrs	No. = 17 AV = 8.1 hrs

NOTE: Some failures affect both the log and linear circuits and are therefore accounted for in both.

TABLE 4-5. EXCORE DETECTOR SAFETY CHANNEL FAILURE PARAMETERS, POSTERIOR DISTRIBUTIONS, AND TOTAL OF INDEPENDENT AND COMMON CAUSE FAILURES

Failure Parameter	Mean	5th Percentile	50th Percentile	95th Percentile
Linear Standby $\lambda_s(hr^{-1})$	3.0-6	1.17-6	2.5-6	5.6-6
Linear Monitored $\lambda_m(hr^{-1})$	3.7-6	1.49-6	3.2-6	6.4-6
Linear Demand $\rho(d^{-1})$	9.7-4	3.2-4	8.3-4	1.66-3
Linear Test-Caused $\Gamma(d^{-1})$	2.4-3	9.3-4	1.91-3	4.5-3
Log Standby $\lambda_s(hr^{-1})$	2.7-6	9.5-7	2.2-6	4.7-6
Log Monitored $\lambda_m(hr^{-1})$	3.7-6	1.49-6	3.2-6	6.4-6
Log Demand $\rho(d^{-1})$	9.9-4	3.5-4	8.6-4	1.69-3
Log Test-Caused $\Gamma(d^{-1})$	3.0-3	1.06-3	2.5-3	5.7-3

NOTE: Exponential notation is indicated in abbreviated form; i.e., 2.9-6 = 2.9×10^{-6}.

required during Mode 6, although detailed calibration is accomplished at that time. This eliminated events that could result from tests and conditions not encountered during normal operations.

A summary of the results of the peer group survey combined with plant-specific data extracted from Table 4-2 is given in Table 4-6. It is important to note that no failures that can be classified as potentially common cause have yet been recorded for the Nuclear Instrument Excore Safety channels at SONGS.

The estimate of the common cause failure parameters must consider the applicability of reported events to the event that this report addresses. This evaluation includes any set of failures that are detected on the same day. Two or more failures observed in this time frame are considered to potentially result from a common cause even though that mechanism was not identified in the root cause analysis. To make the evaluation realistic, these events are weighted by the assessed likelihood that they could have resulted from a common cause mechanism.

In the case of time-related failures (both standby and monitored) of the overpower trip function, the safety channels are providing a continuous reading in the control room. In addition, the daily calimetric calibration check provides a frequent cross reference among the four channels. In order to fail to provide the trip signal, the channels must continue to output signals corresponding to the power output of the reactor and simultaneously be in a state that will prevent them from rising to the trip set point should an overpower transient occur. Failure mechanisms that produce this type of fault are considered unlikely. Hence, the out-of-specification conditions that have been detected in two different channels on the same date as recorded in the peer group data, are assessed to have a 10% probability being due to a double common cause event. As shown in Table 4-6, there are two instances of this condition in the peer group data. When each is assessed as equivalent to 0.1 double common cause failure, the total number of events in the peer group is equivalent to 0.02 double common cause failures, in accordance with the following equation:

$$P\left\{\begin{array}{l}\text{Double Common}\\\text{Cause Failure To}\\\text{Provide Trip Signal}\end{array}\right\} = P\left\{\begin{array}{l}\text{Observation of}\\\text{Two Failures on}\\\text{Same Day is Due}\\\text{to Common Cause}\\\text{Mechanisms}\end{array}\right\} * P\left\{\begin{array}{l}\text{Observed Out-of-}\\\text{Specification}\\\text{Condition Would}\\\text{Produce Failure}\\\text{To Trip, Given}\\\text{Trip Condition}\end{array}\right\}$$

$$= [0.1 * 0.1] * 2 \text{ instances} = 0.02$$

This assessment is considered reasonable because there are no instances at SONGS where two different channels have had observed faults on the same day.

Table 4-7 gives the assumptions and calculations used to estimate the common cause parameter from the above data. The formula estimates beta using both the generic and plant-specific data and is equivalent to a Bayesian update of a noninformative prior by both the Combustion

TABLE 4-6. PEER GROUP SAFETY CHANNEL FAILURE DATA ASSESSMENT OF COMMON CAUSE EVENT POTENTIAL

Unit	Date	Subsystem	Failure Mode	Type Failure	Function Affected	Assessed Failure Likelihood	Time Related Sngl	Time Related Dble	Time Related Trpl	Demand Related Sngl	Demand Related Dble	Demand Related Trpl	Reference	Remarks
Millstone	05/06/88	Ch B 'Neutron Sig'	Fail to Operate	Monitored	Linear	1.00	1.00						NPRDS	
Millstone	09/14/87	Ch C Lin Power	Out of Spec	Standby	Linear	0.10	0.10						NPRDS	
ANO 2	06/03/87	Log Count	Out of Spec	Standby	Log	0.10	0.10						NPRDS	
Palisades	03/07/87	Ch NI-4 Linear Power	Reading Low	Monitored	Linear	0.20	0.20						NPRDS	
Calvert C	10/15/86	Ch B 15 VDC Pwr Sup	Noise in output	Standby	Linear	0.00	0.10						NPRDS	
Calvert C	08/11/86	Lin Power B	Out of Spec	Standby	Linear	0.10	0.10						NPRDS	
Calvert C	08/09/86	Ch D 15 VDC Pwr Sup	Out of Spec	Monitored	Linear	0.10		0.01					NPRDS	Detection Simultaneous with Channel D
Calvert C	08/09/86	Ch C Linear Power	Out of Spec, High	Standby	Linear	0.10							NPRDS	
Calvert C	08/09/86	Ch C 15 VDC Pwr Sup	Out of Spec	Monitored	Linear	0.10							NPRDS	Detection Simultaneous with Channel D
Palisades	05/14/86	Ch NI-6 Power Range	Indication High	Monitored	Linear	0.10	0.10						NPRDS	
Calvert C	04/15/86	Ch A Linear Power	Indication high	Monitored	Linear	0.10	0.10						NPRDS	
Palisades	11/12/85	Ch NI-5 Power Range	Spur. Pretrip Sig	Monitored	Linear	0.05	0.10						NPRDS	
Palisades	11/06/85	Ch NI-5 Power Range	Spurious Trip Sig	Monitored	Linear	0.10	0.10						NPRDS	
Palisades	08/15/85	Ch NI-7 Power Range	Out of Spec	Standby	Linear	0.10		0.01					NPRDS	Detection Simultaneous with NI 7
Palisades	08/15/85	Ch NI-6 Power Range	Out of Spec	Standby	Linear	0.10							NPRDS	Detection Simultaneous with NI 6
Palisades	07/25/85	Ch NI-5 Power Range	Spur Pretrip Sig	Monitored	Linear	0.05	0.05						NPRDS	
Palisades	04/15/85	15 VAC Power Supply	Short to NI 3	Monitored	Log	0.10	0.10						NPRDS	
Palisades	01/29/85	Ch NI-3 Neut. Sensor	Fail to Operate	Monitored	Both	1.00	1.00						NPRDS	
Calvert C	12/14/84	Lin Power C Summer	Erratic Behavior	Test Caused	Linear	0.30				0.30			NPRDS	
Millstone	08/10/84	Ch C Linear Power	Broken Test Pts	Test Caused	N/A	0.00							NPRDS	
Millstone	05/20/84	Ch D Linear Power	Out of Spec	Standby	Linear	0.10	0.10						NPRDS	
Calvert C	04/09/84	Ch C Log Power	Out of Spec, High	Standby	Log	0.10	0.10						NPRDS	
Calvert C	02/02/84	Ch A & B Linear Power	Out of Spec	Standby	Linear	0.10	0.10						NPRDS 840202-1	Not CCF of two channels, should be single failure
Palisades	10/04/83	Ch NI-4 Linear Power	Noisy	Monitored	Both	0.10	0.10						NPRDS	
Palisades	08/22/83	Ch NI-3 Log Sensor	Output Signal High	Monitored	Log	0.10	0.10						NPRDS	
Palisades	05/27/83	Ch NI-5 Power Range	Spurious Trip Sig	Monitored	Linear	0.10	0.10						NPRDS	
Calvert C	05/02/83	Ch 7 Linear Power	Out of Spec	Standby	Linear	0.10	0.10						NPRDS	Both amplifier A & B found drifting
Palisades	01/15/83	Ch NI-3 Log Sensor	Out of Spec	Standby	Log	0.10	0.10						NPRDS	
Palisades	01/12/83	Ch NI-5 Power Range	Fail to Operate	Monitored	Linear	1.00	1.00						NPRDS	
Millstone	11/20/82	Ch D Wide Range NI	Out of Spec	Standby	Log	0.10	0.10						NPRDS	
Millstone	04/06/82	Ch D Wide Range NI	Out of Spec	Standby	Log	0.10	0.10						NPRDS	
Palisades	01/23/82	Wide Range Log Ch	Out of Spec High	Monitored	Log	0.10	0.10						NPRDS	
Calvert C	08/20/81	Lin Power B	Erratic Operation	Monitored	Linear	1.00	1.00						LER 318-81043	
Calvert C	06/01/81	Lin Power A	Fail to Operate	Test Caused	Linear	1.00				1.00			LER 317-81045	
Millstone	05/17/81	Ch D 'Lower' Detector	Indication High	Monitored	Linear	0.05	0.05						NPRDS	
Millstone	04/20/81	Ch D Linear Power	Voltages Unstable	Standby	Linear	0.10	0.10						NPRDS	
Millstone	04/08/81	Ch D Linear Amp	Out of Spec	Standby	Linear	0.10	0.10						NPRDS	
Millstone	04/01/81	Lin Power D	Out of Spec	Standby	Linear	0.10	0.10						LER 336-81016	Repair next shutdown
Palisades	11/09/80	Wide Range Log Ch	Response Delay 30s	Standby	Log	0.50	0.50						NPRDS	
Millstone	10/23/80	Lin Power D	Fail to Operate	Human Error	Linear	1.00				1.00			LER 336-80036	
Millstone	09/30/80	Ch D Wide Range NI	Fail to Operate	Human Error	Log	0.10				0.10			NPRDS	
Millstone	09/08/80	Ch D Wide Range NI	Out of Spec	Standby	Log	0.10	0.10						NPRDS	
Millstone	08/30/80	Ch C Log Power	Out of Spec	Monitored	Log	0.10	0.10						NPRDS	
Palisades	07/03/80	Ch NI-6 Power Range	Unreliable Indic.	Monitored	Linear	0.10	0.10						NPRDS	
Calvert C	12/12/79	Lin Power B	Out of Spec	Standby	Linear	0.10	0.10						LER 318-79044	Channel gain adjusted. MO opened for chamber
Millstone	05/22/79	Lin Power A&D	Fail to Operate	Human Error	Linear	2.00					1.00		LER 336-79012	CCF of two channels, but not STI dependent
Palisades	04/07/79	Wide Range Log Drawer	Output Signal Low	Monitored	Log	1.00	1.00						NPRDS	Unit at 100% power, out of applicable log range
Millstone	04/25/78	Lin Power D	Failed Low	Human Error	Linear	1.00				1.00			LER Unknown	
Palisades	12/01/77	Ch NI-5 Power Range	Fail to Operate	Monitored	Linear	1.00	1.00						NPRDS	
						TOTALS	9.50	0.02		3.40	1.00			

TABLE 4-7. APPLICATION OF MULTIPLE GREEK LETTER MODEL TO OBTAIN COMMON CAUSE FAILURE PARAMETERS (REFERENCE 11)

MGL Parameter Definitions

β ≡ conditional probability that the cause of a component failure will be by one or more additional components.

γ ≡ conditional probability that the cause of a failure that is shared by one or more additional components will be shared by two or more additional components.

n_1, n_2, n_3 ≡ equivalent number of single, double, and triple common cause events, respectively.

MGL Parameter Estimation. The estimated value of β is obtained by combining the plant-specific data and the Combustion Engineering, Inc., data (Reference 2).

	Time-Related			Demand-Related		
	Single	Double	Triple	Single	Double	Triple
SONGS 2 and 3	6.1	0	0	6.1	0	0
Peer Plants	9.5	0.02*	0	3.5	1.1	0
Assessed Events	15.6	0.02	0	9.6	1.1	0

*Includes a 10% probability assessment that simultaneous detection of out-of-specification conditions in two separate channels on the same day results from a common cause event.

$$\beta_T = \frac{2n_2 + 3n_3}{n_1 + 2n_2 + 3n_3} = \frac{2(.02) + 0}{15.6 + 2(.02)}$$

$\beta_T = .0026$

$$\beta_D = \frac{2n_2 + 3n_3}{n_1 + 2n_2 + 3n_3} = \frac{2(1.1) + 0}{9.8 + 2(1.1)}$$

$\beta_D = 0.186$

Although there are no instances of triple failures, there is sufficient experience to warrant including it. Use data for similar systems from a recent PRA (Reference 10).

$\gamma D = \gamma T = 0.07$

Engineering, Inc., and plant-specific data. Since no triple failures have been observed, gamma is taken from data used for similar equipment in a recent PRA (Reference 10).

4.2.4 COMPONENT PARAMETERS

The multiple Greek letter method (Reference 11) used to quantify the contribution of common cause failures in this model is applied to the total failure parameters in Table 4-8. The resulting parameters are point estimates (mean value) of the failure rate. Because FRANTIC and SOCRATES do not have the capability to calculate uncertainty distributions, the distributions developed in this study will be used as a guide to the range over which sensitivity calculations should be accomplished.

4.3 ALTERNATE TESTING POLICIES

Based on scheduling considerations, this analysis addresses testing intervals that vary by increments of 730 hours, which correspond to 1/12 of a year or an average whole month. Two testing policies are addressed:

- Staggered Testing. This policy assumes that the tests of the individual channels are equally spaced in time so that the interval between any two adjacent tests is one-fourth the test interval of an individual test.
- Sequential Testing. This policy assumes that the tests of all four channels are accomplished one after the other, subject to the constraint that no channel shall be bypassed for surveillance testing when another is being repaired.

In practice, a surveillance test schedule will not adhere strictly to either of these policies. However, the calculations show that there is very little difference in the unavailabilities resulting from the two policies.

4.4 RESULTS

System unavailability is evaluated using the SOCRATES computer code (Reference 5). This code has been designed with many of the models contained in FRANTIC, and it has many convenient features for investigating testing policies in support of technical specifications modifications.

To investigate the unavailability implications of extending the surveillance test of the excore nuclear instrumentation safety channel test interval, sensitivity studies are accomplished for the following combinations of conditions:

- Channel standby failure rate at its mean and at the 5th and 95th percentile values (designated by the parameter lambda in SOCRATES output).

TABLE 4-8. SONGS UNIT 2 AND 3 EXCORE SAFETY CHANNEL FAILURE RATE PARAMETERS

Parameter	Application Formula	5th Percentile	Mean	95th Percentile
A. Independent Failures				
"Standby" Lambda	$(1-\beta_T)\lambda_s$	1.17×10^{-6}	2.9×10^{-6}	5.6×10^{-6}
Constant: Monitored + Demand	$(1-\beta_T)\lambda_M T_R + (1-\beta_d)\rho$	2.8×10^{-4}	8.4×10^{-4}	1.44×10^{-3}
Test Caused	$(1-\beta_d)\Gamma$	7.6×10^{-4}	1.96×10^{-3}	3.7×10^{-3}
B. Double Failures				
Standby Failure Rate	$\frac{1}{3}\beta_T(1-\gamma)\lambda_s$	9.0×10^{-9}	2.4×10^{-8}	4.3×10^{-8}
Constant: Monitored + Demand	$\frac{1}{3}\beta_T(1-\gamma)\lambda_M T_R + \frac{1}{3}\beta_D(1-\gamma)\rho$	1.85×10^{-5}	5.6×10^{-5}	9.6×10^{-5}
Test Caused	$\frac{1}{3}\beta_D(1-\gamma)\Gamma$	5.4×10^{-5}	1.38×10^{-4}	2.6×10^{-4}
C. Triple Failures				
Standby Failure Rate	$\beta_T\gamma\lambda_s$	2.0×10^{-9}	5.5×10^{-9}	9.7×10^{-9}
Constant: Monitored + Demand	$\beta_T\gamma\lambda_M T_R + \beta_D\gamma\rho$	4.2×10^{-5}	1.26×10^{-5}	2.4×10^{-5}
Test-caused	$\beta_D\gamma\Gamma$	1.20×10^{-5}	3.1×10^{-5}	5.8×10^{-5}

Common Cause Failure Parameter Calculation. The failure parameter for multiple failures may be obtained from its total failure parameter by the following formula:

$Q_1 = (1 - \beta)\, Q_T$ (single failures)

$Q_2 = \frac{1}{3}\beta(1-\gamma)\, Q_T$ (double failures)

$Q_3 = \beta\gamma Q_T$ (triple failures)

Where Q_T may be the parameter for time-related or demand-related failures.

- Channel bypass time for testing at 1, 2, and 3 hours (designated by the parameter C in SOCRATES output). During testing, the bypassed channel is unavailable to accomplish its safety function, and the trip logic becomes two out of three.

  The best estimate duration of channel bypass for a test is approximately 2 hours. The other bypass times are used to provide a basis to judge the sensitivity of the results to this parameter.

- Surveillance test intervals ranging from 730 hours (1 month) to 4,380 hours (6 months).

- Both staggered and sequential testing policies.

The input echo from the SOCRATES output is given in Table 4-1 for the case using mean failure parameters and staggered testing. The results of the sensitivity studies in terms of average unavailability of the system are given in Table 4-9. This table gives the results from two separate runs. The results for the staggered testing are given at the top of each sheet of the table, and the results for sequential testing with the same set of parameters are given on the bottom of the page. The surveillance test interval is varied from 730 to 4,380 hours in every output table. Sheets 1 to 3 correspond to the mean failure parameters from Table 4-8, with the bypass time rising from 1 hour to 3 hours from sheet 1 to sheet 3. Sheets 4 to 6 and 7 to 9 repeat this process for the 95th and 5th percentiles of the failure parameters, respectively.

From Table 4-9, the following results can be summarized:

1. System unavailability does not change significantly as the test interval varies between 1 and 4 months. For the SONGS base case, a bypass time of 2 hours (the expected duration of a channel bypass for testing) and the mean values of the failure parameters (Table 4-1), the total unavailability declines by about 9% to a minimum as the interval increases from 1 to 3 months and rises by only 1% in the fourth month.

2. System unavailability is relatively insensitive to channel bypass time, increasing slightly and favoring longer test intervals as the bypass time increases.

3. System unavailability is insensitive to a policy of sequential versus staggered testing. The only instance when a 3-month test interval did not produce a minimum unavailability was the case of sequential testing and a lambda of 5.3×10^{-6} per hour, the 95th percentile. For these cases, the minimum occurred at the 2-month interval; however, the unavailability for the 3-month interval was below the current test interval of 1 month. Considering the assumptions used to generate the failure parameters, this variation is judged to be insignificant. Therefore, policies that provide the maximum administrative efficiency and minimize the potential for human error may be selected without worry about the impact of minor scheduling changes.

TABLE 4-9. SOCRATES OUTPUT OF AVERAGE UNAVAILABILITY OF EXCORE NUCLEAR INSTRUMENTATION SAFETY CHANNEL UNDER VARIOUS TEST STRATEGIES
(Failure Parameters per Table 4-8)
(Sheet 1 of 9)

Staggered Testing - Mean Value Parameters, 1-Hour Bypass Time

```
*************************************************************************
* TABLE   1. 1                                                          *
* -----------                                                           *
*                                                                       *
* AVERAGE VALUES AS A FUNCTION OF TEST INTERVAL                         *
* FOR COMPONENT GROUP    1  COMPONENTS OR TESTS: TESTA     , TESTB    , *
*                                                TESTC     , TESTD      *
*************************************************************************
*                                                                       *
* PARAMETERS CHANGED AND HELD FIXED IN THIS TABLE                       *
* ---------------------------------------------                         *
*     COMPONENT GROUP    1,          C=1.000E+00                        *
*                                                                       *
*-----------+---------+---------+---------+---------+
*   TABLE   +DOWNTIME +TESTTIME +BETWN TST+  TOTAL  +
* VARIABLE- + CONTRIB + CONTRIB + CONTRIB + CONTRIB +
*   CGROUP  +         +         +         +         +
* T     - 1 +         +         +         +         +
*-----------+---------+---------+---------+---------+
*    730.      1.33 -5   1.41 -6   4.89 -5   6.36 -5
*   1.460E+03  6.74 -6   7.56 -7   5.16 -5   5.91 -5
*   2.190E+03  4.56 -6   5.49 -7   5.33 -5   5.84 -5
*   2.920E+03  3.47 -6   4.55 -7   5.48 -5   5.88 -5
*   3.650E+03  2.82 -6   4.05 -7   5.64 -5   5.96 -5
*   4.380E+03  2.39 -6   3.78 -7   5.81 -5   6.08 -5
*****************************
```

Sequential Testing - Mean Value Parameters, 1-Hour Bypass Time

```
*************************************************************************
* TABLE   1. 1                                                          *
* -----------                                                           *
*                                                                       *
* AVERAGE VALUES AS A FUNCTION OF TEST INTERVAL                         *
* FOR COMPONENT GROUP    1  COMPONENTS OR TESTS: TESTA     , TESTB    , *
*                                                TESTC     , TESTD      *
*************************************************************************
*                                                                       *
* PARAMETERS CHANGED AND HELD FIXED IN THIS TABLE                       *
* ---------------------------------------------                         *
*     COMPONENT GROUP    1,          C=1.000E+00                        *
*                                                                       *
*-----------+---------+---------+---------+---------+
*   TABLE   +DOWNTIME +TESTTIME +BETWN TST+  TOTAL  +
* VARIABLE- + CONTRIB + CONTRIB + CONTRIB + CONTRIB +
*   CGROUP  +         +         +         +         +
* T     - 1 +         +         +         +         +
*-----------+---------+---------+---------+---------+
*    730.      1.33 -5   1.21 -6   4.94 -5   6.40 -5
*   1.460E+03  6.78 -6   6.67 -7   5.26 -5   6.01 -5
*   2.190E+03  4.60 -6   5.02 -7   5.50 -5   6.01 -5
*   2.920E+03  3.51 -6   4.33 -7   5.74 -5   6.13 -5
*   3.650E+03  2.87 -6   4.01 -7   5.99 -5   6.32 -5
*   4.380E+03  2.44 -6   3.89 -7   6.28 -5   6.56 -5
*****************************
```

TABLE 4-9. SOCRATES OUTPUT OF AVERAGE UNAVAILABILITY OF EXCORE NUCLEAR INSTRUMENTATION SAFETY CHANNEL UNDER VARIOUS TEST STRATEGIES
(Failure Parameters per Table 4-8)
(Sheet 2 of 9)

Staggered Testing - Mean Value Parameters, 2-Hour Bypass Time

```
*****************************************************************************
*  TABLE   1. 2                                                             *
*  ------------                                                             *
*                                                                           *
*  AVERAGE VALUES AS A FUNCTION OF TEST INTERVAL                            *
*  FOR COMPONENT GROUP    1  COMPONENTS OR TESTS: TESTA     , TESTB      ,  *
*                                                 TESTC     , TESTD         *
*****************************************************************************
*                                                                           *
*  PARAMETERS CHANGED AND HELD FIXED IN THIS TABLE                          *
*  ---------------------------------------------                            *
*      COMPONENT GROUP    1,          C=2.000E+00                           *
*                                                                           *
*-----------+----------+----------+----------+----------+
*   TABLE   +DOWNTIME  +TESTTIME  +BETWN TST+  TOTAL    +
* VARIABLE- + CONTRIB  + CONTRIB  + CONTRIB  + CONTRIB  +
*   CGROUP  +          +          +          +          +
* T     - 1 +          +          +          +          +
*-----------+----------+----------+----------+----------+
*    730.     1.33 -5   2.83 -6   4.87 -5   6.48 -5
*   1.460E+03 6.74 -6   1.51 -6   5.15 -5   5.97 -5
*   2.190E+03 4.56 -6   1.10 -6   5.32 -5   5.89 -5
*   2.920E+03 3.47 -6   9.09 -7   5.48 -5   5.92 -5
*   3.650E+03 2.82 -6   8.10 -7   5.64 -5   6.00 -5
*   4.380E+03 2.39 -6   7.56 -7   5.80 -5   6.12 -5
*****************************
```

Sequential Testing - Mean Value Parameters, 2-Hour Bypass Time

```
*****************************************************************************
*  TABLE   1. 2                                                             *
*  ------------                                                             *
*                                                                           *
*  AVERAGE VALUES AS A FUNCTION OF TEST INTERVAL                            *
*  FOR COMPONENT GROUP    1  COMPONENTS OR TESTS: TESTA     , TESTB      ,  *
*                                                 TESTC     , TESTD         *
*****************************************************************************
*                                                                           *
*  PARAMETERS CHANGED AND HELD FIXED IN THIS TABLE                          *
*  ---------------------------------------------                            *
*      COMPONENT GROUP    1,          C=2.000E+00                           *
*                                                                           *
*-----------+----------+----------+----------+----------+
*   TABLE   +DOWNTIME  +TESTTIME  +BETWN TST+  TOTAL    +
* VARIABLE- + CONTRIB  + CONTRIB  + CONTRIB  + CONTRIB  +
*   CGROUP  +          +          +          +          +
* T     - 1 +          +          +          +          +
*-----------+----------+----------+----------+----------+
*    730.     1.33 -5   2.42 -6   4.93 -5   6.51 -5
*   1.460E+03 6.78 -6   1.33 -6   5.26 -5   6.07 -5
*   2.190E+03 4.60 -6   1.00 -6   5.50 -5   6.06 -5
*   2.920E+03 3.51 -6   8.65 -7   5.73 -5   6.17 -5
*   3.650E+03 2.87 -6   8.02 -7   5.99 -5   6.35 -5
*   4.380E+03 2.44 -6   7.78 -7   6.27 -5   6.60 -5
*****************************
```

TABLE 4-9. SOCRATES OUTPUT OF AVERAGE UNAVAILABILITY OF EXCORE NUCLEAR INSTRUMENTATION SAFETY CHANNEL UNDER VARIOUS TEST STRATEGIES
(Failure Parameters per Table 4-8)
(Sheet 3 of 9)

Staggered Testing - Mean Value Parameters, 3-Hour Bypass Time

TABLE 1. 3

AVERAGE VALUES AS A FUNCTION OF TEST INTERVAL
FOR COMPONENT GROUP 1 COMPONENTS OR TESTS: TESTA , TESTB , TESTC , TESTD

PARAMETERS CHANGED AND HELD FIXED IN THIS TABLE

COMPONENT GROUP 1, C=3.000E+00

TABLE VARIABLE- CGROUP T - 1	DOWNTIME CONTRIB	TESTTIME CONTRIB	BETWN TST CONTRIB	TOTAL CONTRIB
730.	1.33 -5	4.24 -6	4.85 -5	6.60 -5
1.460E+03	6.74 -6	2.27 -6	5.13 -5	6.04 -5
2.190E+03	4.56 -6	1.65 -6	5.31 -5	5.93 -5
2.920E+03	3.47 -6	1.36 -6	5.47 -5	5.96 -5
3.650E+03	2.82 -6	1.22 -6	5.63 -5	6.03 -5
4.380E+03	2.39 -6	1.13 -6	5.80 -5	6.15 -5

Sequential Testing - Mean Value Parameters, 3-Hour Bypass Time

TABLE 1. 3

AVERAGE VALUES AS A FUNCTION OF TEST INTERVAL
FOR COMPONENT GROUP 1 COMPONENTS OR TESTS: TESTA , TESTB , TESTC , TESTD

PARAMETERS CHANGED AND HELD FIXED IN THIS TABLE

COMPONENT GROUP 1, C=3.000E+00

TABLE VARIABLE- CGROUP T - 1	DOWNTIME CONTRIB	TESTTIME CONTRIB	BETWN TST CONTRIB	TOTAL CONTRIB
730.	1.33 -5	3.63 -6	4.92 -5	6.62 -5
1.460E+03	6.78 -6	2.00 -6	5.25 -5	6.13 -5
2.190E+03	4.60 -6	1.51 -6	5.49 -5	6.10 -5
2.920E+03	3.51 -6	1.30 -6	5.73 -5	6.21 -5
3.650E+03	2.87 -6	1.20 -6	5.98 -5	6.39 -5
4.380E+03	2.44 -6	1.17 -6	6.27 -5	6.63 -5

TABLE 4-9. SOCRATES OUTPUT OF AVERAGE UNAVAILABILITY OF EXCORE NUCLEAR INSTRUMENTATION SAFETY CHANNEL UNDER VARIOUS TEST STRATEGIES
(Failure Parameters per Table 4-8)
(Sheet 4 of 9)

Staggered Testing - 95th Percentile Parameters, 1-Hour Bypass Time

TABLE 1. 1

AVERAGE VALUES AS A FUNCTION OF TEST INTERVAL
FOR COMPONENT GROUP 1 COMPONENTS OR TESTS: TESTA , TESTB , TESTC , TESTD

PARAMETERS CHANGED AND HELD FIXED IN THIS TABLE

COMPONENT GROUP 1, C=1.000E+00

TABLE VARIABLE- CGROUP T - 1	DOWNTIME CONTRIB	TESTTIME CONTRIB	BETWN TST CONTRIB	TOTAL CONTRIB
730.	2.52 -5	2.48 -6	9.32 -5	1.21 -4
1.460E+03	1.29 -5	1.35 -6	9.93 -5	1.14 -4
2.190E+03	8.80 -6	1.02 -6	1.04 -4	1.14 -4
2.920E+03	6.77 -6	8.86 -7	1.08 -4	1.16 -4
3.650E+03	5.57 -6	8.29 -7	1.13 -4	1.20 -4
4.380E+03	4.79 -6	8.11 -7	1.19 -4	1.24 -4

Sequential Testing - 95th Percentile Parameters, 1-Hour Bypass Time

TABLE 1. 1

AVERAGE VALUES AS A FUNCTION OF TEST INTERVAL
FOR COMPONENT GROUP 1 COMPONENTS OR TESTS: TESTA , TESTB , TESTC , TESTD

PARAMETERS CHANGED AND HELD FIXED IN THIS TABLE

COMPONENT GROUP 1, C=1.000E+00

TABLE VARIABLE- CGROUP T - 1	DOWNTIME CONTRIB	TESTTIME CONTRIB	BETWN TST CONTRIB	TOTAL CONTRIB
730.	2.52 -5	2.10 -6	9.43 -5	1.22 -4
1.460E+03	1.30 -5	1.20 -6	1.02 -4	1.16 -4
2.190E+03	8.88 -6	9.66 -7	1.08 -4	1.18 -4
2.920E+03	6.87 -6	8.90 -7	1.15 -4	1.23 -4
3.650E+03	5.68 -6	8.80 -7	1.23 -4	1.30 -4
4.380E+03	4.92 -6	9.01 -7	1.34 -4	1.40 -4

TABLE 4-9. SOCRATES OUTPUT OF AVERAGE UNAVAILABILITY OF EXCORE NUCLEAR INSTRUMENTATION SAFETY CHANNEL UNDER VARIOUS TEST STRATEGIES
(Failure Parameters per Table 4-8)
(Sheet 5 of 9)

Staggered Testing - 95th Percentile Parameters, 2-Hour Bypass Time

TABLE 1. 2

AVERAGE VALUES AS A FUNCTION OF TEST INTERVAL
FOR COMPONENT GROUP 1 COMPONENTS OR TESTS: TESTA , TESTB , TESTC , TESTD

PARAMETERS CHANGED AND HELD FIXED IN THIS TABLE

COMPONENT GROUP 1, C=2.000E+00

TABLE VARIABLE- CGROUP T - 1	DOWNTIME CONTRIB	TESTTIME CONTRIB	BETWN TST CONTRIB	TOTAL CONTRIB
730.	2.52 -5	4.95 -6	9.28 -5	1.23 -4
1.460E+03	1.29 -5	2.71 -6	9.91 -5	1.15 -4
2.190E+03	8.80 -6	2.04 -6	1.04 -4	1.14 -4
2.920E+03	6.77 -6	1.77 -6	1.08 -4	1.17 -4
3.650E+03	5.57 -6	1.66 -6	1.13 -4	1.20 -4
4.380E+03	4.79 -6	1.62 -6	1.19 -4	1.25 -4

Sequential Testing - 95th Percentile Parameters, 2-Hour Bypass Time

TABLE 1. 2

AVERAGE VALUES AS A FUNCTION OF TEST INTERVAL
FOR COMPONENT GROUP 1 COMPONENTS OR TESTS: TESTA , TESTB , TESTC , TESTD

PARAMETERS CHANGED AND HELD FIXED IN THIS TABLE

COMPONENT GROUP 1, C=2.000E+00

TABLE VARIABLE- CGROUP T - 1	DOWNTIME CONTRIB	TESTTIME CONTRIB	BETWN TST CONTRIB	TOTAL CONTRIB
730.	2.52 -5	4.20 -6	9.41 -5	1.23 -4
1.460E+03	1.30 -5	2.41 -6	1.01 -4	1.17 -4
2.190E+03	8.88 -6	1.[illegible]3 -6	1.08 -4	1.19 -4
2.920E+03	6.87 -6	1.78 -6	1.15 -4	1.23 -4
3.650E+03	5.68 -6	1.76 -6	1.23 -4	1.31 -4
4.380E+03	4.92 -6	1.8[illegible] -6	1.34 -4	1.41 -4

TABLE 4-9. SOCRATES OUTPUT OF AVERAGE UNAVAILABILITY OF EXCORE NUCLEAR INSTRUMENTATION SAFETY CHANNEL UNDER VARIOUS TEST STRATEGIES
(Failure Parameters per Table 4-8)
(Sheet 6 of 9)

Staggered Testing - 95th Percentile Parameters, 3-Hour Bypass Time

TABLE 1. 3

AVERAGE VALUES AS A FUNCTION OF TEST INTERVAL
FOR COMPONENT GROUP 1 COMPONENTS OR TESTS: TESTA , TESTB , TESTC , TESTD

PARAMETERS CHANGED AND HELD FIXED IN THIS TABLE

COMPONENT GROUP 1, C=3.000E+00

TABLE VARIABLE- CGROUP T - 1	DOWNTIME CONTRIB	TESTTIME CONTRIB	BETWN TST CONTRIB	TOTAL CONTRIB
730.	2.52 -5	7.43 -6	9.24 -5	1.25 -4
1.460E+03	1.29 -5	4.06 -6	9.89 -5	1.16 -4
2.190E+03	8.80 -6	3.06 -6	1.03 -4	1.15 -4
2.920E+03	6.77 -6	2.66 -6	1.08 -4	1.17 -4
3.650E+03	5.57 -6	2.49 -6	1.13 -4	1.21 -4
4.380E+03	4.79 -6	2.43 -6	1.19 -4	1.26 -4

Sequential Testing - 95th Percentile Parameters, 3-Hour Bypass Time

TABLE 1. 3

AVERAGE VALUES AS A FUNCTION OF TEST INTERVAL
FOR COMPONENT GROUP 1 COMPONENTS OR TESTS: TESTA , TESTB , TESTC , TESTD

PARAMETERS CHANGED AND HELD FIXED IN THIS TABLE

COMPONENT GROUP 1, C=3.000E+00

TABLE VARIABLE- CGROUP T - 1	DOWNTIME CONTRIB	TESTTIME CONTRIB	BETWN TST CONTRIB	TOTAL CONTRIB
730.	2.52 -5	6.30 -6	9.38 -5	1.25 -4
1.460E+03	1.30 -5	3.61 -6	1.01 -4	1.18 -4
2.190E+03	8.88 -6	2.89 -6	1.08 -4	1.19 -4
2.920E+03	6.87 -6	2.67 -6	1.15 -4	1.24 -4
3.650E+03	5.68 -6	2.64 -6	1.23 -4	1.32 -4
4.380E+03	4.92 -6	2.70 -6	1.34 -4	1.41 -4

TABLE 4-9. SOCRATES OUTPUT OF AVERAGE UNAVAILABILITY OF EXCORE NUCLEAR INSTRUMENTATION SAFETY CHANNEL UNDER VARIOUS TEST STRATEGIES
(Failure Parameters per Table 4-8)
(Sheet 7 of 9)

Staggered Testing - 5th Percentile Parameters, 1-Hour Bypass Time

TABLE 1. 1

AVERAGE VALUES AS A FUNCTION OF TEST INTERVAL
FOR COMPONENT GROUP 1 COMPONENTS OR TESTS: TESTA , TESTB , TESTC , TESTD

PARAMETERS CHANGED AND HELD FIXED IN THIS TABLE

COMPONENT GROUP 1, C=1.000E+00

TABLE VARIABLE- CGROUP T - 1	DOWNTIME CONTRIB	TESTTIME CONTRIB	BETWN TST CONTRIB	TOTAL CONTRIB
730.	4.74 -6	4.56 -7	1.60 -5	2.12 -5
1.460E+03	2.39 -6	2.36 -7	1.68 -5	1.94 -5
2.190E+03	1.60 -6	1.65 -7	1.72 -5	1.89 -5
2.920E+03	1.21 -6	1.31 -7	1.75 -5	1.88 -5
3.650E+03	9.77 -7	1.11 -7	1.77 -5	1.88 -5
4.380E+03	8.20 -7	9.93 -8	1.80 -5	1.89 -5

Sequential Testing - 5th Percentile Parameters, 1-Hour Bypass Time

TABLE 1. 1

AVERAGE VALUES AS A FUNCTION OF TEST INTERVAL
FOR COMPONENT GROUP 1 COMPONENTS OR TESTS: TESTA , TESTB , TESTC , TESTD

PARAMETERS CHANGED AND HELD FIXED IN THIS TABLE

COMPONENT GROUP 1, C=1.000E+00

TABLE VARIABLE- CGROUP T - 1	DOWNTIME CONTRIB	TESTTIME CONTRIB	BETWN TST CONTRIB	TOTAL CONTRIB
730.	4.76 -6	3.88 -7	1.62 -5	2.14 -5
1.460E+03	2.40 -6	2.04 -7	1.71 -5	1.97 -5
2.190E+03	1.62 -6	1.46 -7	1.77 -5	1.95 -5
2.920E+03	1.23 -6	1.18 -7	1.82 -5	1.96 -5
3.650E+03	9.91 -7	1.03 -7	1.87 5	1.98 -5
4.380E+03	8.35 -7	9.49 -8	1.92 -5	2.02 -5
5.110E+03	7.23 -7	8.99 -8	1.98 -5	2.06 -5
5.840E+03	6.40 -7	8.72 -8	2.03 -5	2.11 -5
6.380E+03	5.91 -7	8.61 -8	2.08 -5	2.14 -5

TABLE 4-9. SOCRATES OUTPUT OF AVERAGE UNAVAILABILITY OF EXCORE NUCLEAR INSTRUMENTATION SAFETY CHANNEL UNDER VARIOUS TEST STRATEGIES
(Failure Parameters per Table 4-8)
(Sheet 8 of 9)

Staggered Testing - 5th Percentile Parameters, 2-Hour Bypass Time

TABLE 1. 2

AVERAGE VALUES AS A FUNCTION OF TEST INTERVAL
FOR COMPONENT GROUP 1 COMPONENTS OR TESTS: TESTA , TESTB , TESTC , TESTD

PARAMETERS CHANGED AND HELD FIXED IN THIS TABLE

COMPONENT GROUP 1, C=2.000E+00

TABLE VARIABLE- CGROUP T - 1	DOWNTIME CONTRIB	TESTTIME CONTRIB	BETWN TST CONTRIB	TOTAL CONTRIB
730.	4.75 -6	9.11 -7	1.60 -5	2.16 -5
1.460E+03	2.39 -6	4.73 -7	1.67 -5	1.96 -5
2.190E+03	1.61 -6	3.30 -7	1.71 -5	1.91 -5
2.920E+03	1.21 -6	2.62 -7	1.74 -5	1.89 -5
3.650E+03	9.77 -7	2.23 -7	1.77 -5	1.89 -5
4.380E+03	8.20 -7	1.99 -7	1.80 -5	1.90 -5

Sequential Testing - 5th Percentile Parameters, 2-Hour Bypass Time

TABLE 1. 2

AVERAGE VALUES AS A FUNCTION OF TEST INTERVAL
FOR COMPONENT GROUP 1 COMPONENTS OR TESTS: TESTA , TESTB , TESTC , TESTD

PARAMETERS CHANGED AND HELD FIXED IN THIS TABLE

COMPONENT GROUP 1, C=2.000E+00

TABLE VARIABLE- CGROUP T - 1	DOWNTIME CONTRIB	TESTTIME CONTRIB	BETWN TST CONTRIB	TOTAL CONTRIB
730.	4.77 -6	7.75 -7	1.62 -5	2.17 -5
1.460E+03	2.41 -6	4.08 -7	1.71 -5	1.99 -5
2.190E+03	1.[illegible] -6	2.91 -7	1.77 -5	1.96 -5
2.920E+03	1.23 -6	2.36 -7	1.82 -5	1.97 -5
3.650E+03	9.92 -7	2.07 -7	1.87 -5	1.99 -5
4.380E+03	8.36 -7	1.90 -7	1.92 -5	2.03 -5
5.110E+03	7.24 -7	1.80 -7	1.98 -5	2.07 -5
5.840E+03	6.41 -7	1.74 -7	2.03 -5	2.11 -5
6.380E+03	5.91 -7	1.72 -7	2.08 -5	2.15 -5

TABLE 4-9. SOCRATES OUTPUT OF AVERAGE UNAVAILABILITY OF EXCORE NUCLEAR INSTRUMENTATION SAFETY CHANNEL UNDER VARIOUS TEST STRATEGIES
(Failure Parameters per Table 4-8)
(Sheet 9 of 9)

Staggered Testing - 5th Percentile Parameters, 3-Hour Bypass Time

TABLE 1. 3

AVERAGE VALUES AS A FUNCTION OF TEST INTERVAL
FOR COMPONENT GROUP 1 COMPONENTS OR TESTS: TESTA, TESTB, TESTC, TESTD

PARAMETERS CHANGED AND HELD FIXED IN THIS TABLE

COMPONENT GROUP 1, C=3.000E+00

TABLE VARIABLE- CGROUP T - 1	DOWNTIME CONTRIB	TESTTIME CONTRIB	BETWN TST CONTRIB	TOTAL CONTRIB
730.	4.75 -6	1.37 -6	1.59 -5	2.20 -5
1.460E+03	2.39 -6	7.09 -7	1.67 -5	1.98 -5
2.190E+03	1.61 -6	4.95 -7	1.71 -5	1.92 -5
2.920E+03	1.21 -6	3.92 -7	1.74 -5	1.90 -5
3.650E+03	9.77 -7	3.34 -7	1.77 -5	1.90 -5
4.380E+03	8.21 -7	2.98 -7	1.80 -5	1.91 -5

Sequential Testing - 5th Percentile Parameters, 3-Hour Bypass Time

TABLE 1. 3

AVERAGE VALUES AS A FUNCTION OF TEST INTERVAL
FOR COMPONENT GROUP 1 COMPONENTS OR TESTS: TESTA, TESTB, TESTC, TESTD

PARAMETERS CHANGED AND HELD FIXED IN THIS TABLE

COMPONENT GROUP 1, C=3.000E+00

TABLE VARIABLE- CGROUP T - 1	DOWNTIME CONTRIB	TESTTIME CONTRIB	BETWN TST CONTRIB	TOTAL CONTRIB
730.	4.76 -6	1.16 -6	1.62 -5	2.21 -5
1.460E+03	2.41 -6	6.12 -7	1.71 -5	2.01 -5
2.190E+03	1.62 -6	4.36 -7	1.77 -5	1.97 -5
2.920E+03	1.23 -6	3.54 -7	1.82 -5	1.98 -5
3.650E+03	9.92 -7	3.10 -7	1.87 -5	2.00 -5
4.380E+03	8.35 -7	2.84 -7	1.92 -5	2.04 -5
5.110E+03	7.24 -7	2.70 -7	1.98 -5	2.08 -5
5.840E+03	6.40 -7	2.61 -7	2.03 -5	2.12 -5
6.380E+03	5.91 -7	2.58 -7	2.07 -5	2.16 -5

Given the above arguments, it is reasonable to conclude that the quantitative evaluation supports an extension of the excore nuclear instrumentation safety channel test interval from its current 1-month interval to 3 months.

5. CONCLUSIONS AND RECOMMENDATIONS

Conclusions and recommendations regarding the 31-day excore nuclear instrumentation safety channel drawer surveillance test are organized into five areas:

- Reduction of test content.
- Use of test circuits designed into the system.
- Use of operations procedures to satisfy startup test requirements for the log high power trip.
- Consolidation of monthly requirements into the PPS 31-day test.
- Extension of the surveillance test interval.

These areas will be addressed in turn.

5.1 REDUCTION OF TEST CONTENT

The risk-based evaluation of the content and effectiveness of the 31-day excore nuclear instrumentation safety channel drawer test described in Section 3.3 indicated that the following portions of the test may be deleted without affecting safety functions.

- Power Supply Tests. A support system whose proper functioning will be reflected in the proper voltages of the amplifiers. There have been no failures to trip as a result of out-of-specification power supply voltages. Catastrophic failures will be annunciated in the control room. This eliminates one of the sections of the test that requires opening the safety channel drawer.
- Log Channel Functional Test. The monthly requirement is currently satisfied by the PPS 31-day test. This recommendation eliminates duplication.
- $10^{-4\%}$ and 55% Bistable Setpoint Tests. Both activate trip functions but do not generate the trips. Trip function activation is annunciated in the control room. The exact power level is not critical for either safety function. In addition, no failures of these components have been observed during the entire operating history of the reactors.

5.2 USE OF TEST CIRCUITS DESIGNED INTO THE SYSTEM

The risk-based evaluation of the content and effectiveness of the 31-day excore nuclear instrumentation safety channel drawer test described in Section 3.3 indicated that the following portions of the test can be modified to be accomplished from the front panel.

- The rate channel test currently in the procedure can be replaced with a functional check using the rate trip test potentiometer on the front panel. The rate channel is part of the log channel and does not require a monthly calibration. Its alarm is effective primarily during startup and has little safety impact at operating power levels.

- The calibration and functional test requirements of the linear channels can be accomplished using the calibration circuit provided on the front panel. The equivalence of this circuit to a known input was demonstrated in Section 3.3.5, thus satisfying the technical specification requirements for a channel calibration test.

5.3 HIGH LOGARITHMIC POWER TEST REQUIREMENT PRIOR TO STARTUP

The technical specifications require that only a functional test of the high logarithmic power trip is required for both the monthly test and the startup test. Section 3.3 recommends that this requirement be satisfied by a functional test of the high power log trip using the test potentiometer on the front panel, rather than a repeat of the entire test. This eliminates the need to pull the safety channel drawer. The resulting functional test can be easily accomplished within the startup operations procedure or with an abbreviated startup functional test.

5.4 CONSOLIDATION OF MONTHLY REQUIREMENTS INTO THE PPS 31-DAY TEST

The recommendations for the monthly excore safety channel test may be implemented by modifying the existing procedure so that it can be accomplished without opening the safety channel drawer. The result would be a much smaller procedure, but the significant administrative burden of setup, coordination, review, and record keeping discussed in Section 3.2.3 would remain approximately the same.

The discussions in Section 3.3 recommend consolidating the remaining steps into the PPS 31-day test. This has the disadvantage of making the scope of the PPS test broader than originally intended and extending an already very lengthy test. However, it would eliminate SO23-II-5.5(-8) and its associated administrative burdens.

5.5 EXTENSION OF THE SURVEILLANCE TEST INTERVAL

The quantitative evaluation presented in Section 4 supports extending the test interval of the excore nuclear instrumentation safety channels to 92 days. The use of site-specific data to update the more generic failure parameters used in Reference 2 resulted in a system unavailability that is relatively insensitive to the test interval, with system unavailability remaining approximately the same and declining as the test interval increases to 92 days for best estimate failure rates. This result is reasonable since the excore nuclear instrumentation safety channel is an active system in which most catastrophic failures will be revealed when they occur.

The failure data indicated that there is no basis within the failure history of the system to indicate that the logarithmic high power function needs to be tested within 7 days of startup of the reactor.

However, given that the high log power trip will be one of the primary safety trips during startup, including the functional test of the log channel in the startup procedure may be prudent.

5.6 GENERAL

Two additional general conclusions regarding the use of risk-based methods of evaluating surveillance tests can be made. First, the qualitative evaluation of test procedures versus safety functions provides valuable insights into system operation and the effect of technical specification requirements on risk. It points to areas of duplication and unnecessary detail that can be modified or eliminated. Second, the data evaluation provides insights into test effectiveness and input for failure parameters. This insight can be important for both the qualitative and the quantitative analysis.

6. REFERENCES

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4. Ginzburg, T., J. L. Boccio, and R. E. Hall, "FRANTIC II: Applications to Standby Safety Systems," Brookhaven National Laboratory, prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-3627, BNL-NUREG-51738, December 1983.

5. Electric Power Research Institute, "PC SOCRATES Version 1.02 User's Guide," draft report, September 2, 1987.

6. Southern California Edison Company, "Excore Nuclear Instrumentation System," SONGS 2 and 3 System Description SD-SO23-470, Revision 0.

7. General Atomic Vendor Safety Channel Operation and Maintenance Manual; SO23-941-45-13.

8. U.S. Nuclear Regulatory Commission, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400, NUREG-75/014, October 1975.

9. IEEE Guide to the Collection and Presentation of Electrical and Sensing Stations, IEEE-STD500-1977.

10. Pickard, Lowe and Garrick, Inc., "Fermi 2 Level 1 Probabilistic Risk Assessment," Interim Report, PLG-0676, January 1989.

11. Pickard, Lowe and Garrick, Inc., "PRA Procedures for Dependent Events Analysis, Volume II - System Level Analysis," PLG-0453, December 1985.

APPENDIX A

SONGS UNIT 2 TECHNICAL SPECIFICATIONS
TABLE 4.3-1, REACTOR PROTECTIVE INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	#	1, 2, 3*, 4*, 5*
2. Linear Power Level - High	S	D(2,4), M(3,4), Q(4), #(4)	M	1, 2
3. Logarithmic Power Level - High	S	#(4)	M and S/U(1)	1, 2, 3, 4, 5
4. Pressurizer Pressure - High	S	#	M	1, 2
5. Pressurizer Pressure - Low	S	#	M	1, 2
6. Containment Pressure - High	S	#	M	1, 2
7. Steam Generator Pressure - Low	S	#	M	1, 2
8. Steam Generator Level - Low	S	#	M	1, 2
9. Local Power Density - High	S	D(2,4), #(4,5)	M, #(6)	1, 2
10. DNBR - Low	S	S(7), D(2,4), M(8), #(4,5)	M, #(6)	1, 2
11. Steam Generator Level - High	S	#	M	1, 2
12. Reactor Protection System Logic	N.A.	N.A.	M	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Reactor Trip Breakers	N.A.	N.A.	M,(12)	1, 2, 3*, 4*, 5*
14. Core Protection Calculators	S	D(2,4),S(7) #(4,5),M(8)	M(11),#(6)	1, 2
15. CEA Calculators	S	#	M,#(6)	1, 2
16. Reactor Coolant Flow-Low	S	#	M	1, 2
17. Seismic-High	S	#	M	1, 2
18. Loss of Load	S	N.A.	M	1 (9)

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

- At least once per Refueling Interval.

(1) - Each startup or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust the Linear Power Level signals and the CPC addressable constant multipliers to make the CPC delta T power and CPC nuclear power calculations agree with the calorimetric calculation if absolute difference is greater than 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation (conservatively compensated for measurement uncertainties) or by calorimetric calculations (conservatively compensated for measurement uncertainties) and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations (conservatively compensated for measurement uncertainties).

(9) - Above 55% of RATED THERMAL POWER.

(10) - Deleted.

TABLE 4.3-1 (Continued)

TABLE NOTATION

(11) - The monthly CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC.

(12) - At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage and shunt trips.

APPENDIX B

TECHNICAL SPECIFICATION SURVEILLANCES ON EXCORE SAFETY CHANNELS AND RELATED EQUIPMENT

A brief description and an outline of the applicable sections (as required) is provided for surveillance tests that verify the operability of the same portion of the excore nuclear instrumentation safety channels.

I. SO23-II-5.5 through SO23-II-5.8, Revision 10

A. Title. Nuclear instrumentation safety channel A through D drawer test - linear power subchannel gains - channel functional test and channel calibration (31-day interval; startup).

B. Description. Nuclear instrumentation monthly functional test and channel calibration specifically for the safety channel drawer itself. This test is also performed for each channel prior to every reactor startup.

C. Responsible Group. Station instrumentation and control.

D. Outline

1. Section 6.1 - Setup

2. Section 6.2.1 - Power Supply Check

3. Section 6.2.2 - Logarithmic Circuits

4. Section 6.2.3 - 10^{-4} Bistable

5. Section 6.2.4 - Rate Channel

6. Section 6.2.5 through Section 6.2.8 - Linear Amplifiers A10, A11, A12

7. Section 6.2.9 - Summer and Op. Amp A13 and Isolation Buffer A15

8. Section 6.2.10 - 55% Bistable

9. Section 6.2.11 - CPC Reset

10. Section 6.2.12 - Steam Generator Low Flow Bypass Reset

II. S023-II-5.1 through S023-II-5.4, Revision 6

A. Title Nuclear Instrumentation Safety Channel Drawer - Logarithmic Power and Linear Power Level Channel Calibration (18-month interval).

B. Description. Nuclear instrumentation 18-month calibration check for which an extensive calibration on the power supplies, linear and log power circuitry, and bistable are performed.

C. Responsible Group. Station Instrumentation and Control.

D. Outline

1. Section 6.1 through Section 6.5 - Setup
2. Section 6.6 - Power Supply PS-1 ($\pm$15V)
3. Section 6.7 - Power Supply PS-2 (H.V)
4. Section 6.8 - Tennelec Pulser Setup
5. Section 6.9 - Calibrator and Signal Selector Calibration
6. Section 6.10 - Logarithmic Count Rate Discriminator Threshold
7. Section 6.11 - Logarithmic Count Rate Circuitry
8. Section 6.12 - Calibration Signal Selector
9. Section 6.13 - Logarithmic Campbell Circuitry
10. Section 6.14 - Alignment Check - Wide-Range Logarithmic Power Channel
11. Section 6.15 - Period Amplifier A7 - Rate Meter Calibration
12. Section 6.16 - Linear Amplifier A10
13. Section 6.17 - Linear Amplifier A11
14. Section 6.18 - Linear Amplifier A12
15. Section 6.19 - Summer and Optional Amplifier A13 and Isolation Buffer A15
16. Section 6.20 - Isolation Buffer A14
17. Section 6.21 - 10^{-4} Bistable A16 Test

18. Section 6.22 - 55% Bistable A17 Test

19. Section 6.23 - Bistable Trip A18 "Trouble"

III. SO23-II-1.1.1 through SO23-II-1.1.4, Revision 0

A. Title. Reactor Plant Protection System, Channel A through D, Channel Functional Test (31-day interval).

B. Description. PPS 31-day functional test that verifies operation of all the trip functions and other circuitry setpoints (i.e., annunciators, test circuitry, etc.) for the PPS.

C. Responsible Group. Station Instrumentation and Control.

D. Outline

1. Section 6.1 - Power Supply Test

2. Section 6.2 - Ground Detector Test

3. Section 6.3 - Bistable Comparator and Variable Setpoint Lamp Test

4. Section 6.4 - Bistable Control Panel Digital Voltmeter Test

5. Section 6.5 - Initial Setup

6. Section 6.6 - High Linear Power Level

7. Section 6.7 - Loss of Load Trip

8. Section 6.8 - 10^{-4} Bistable Interface Test

9. Section 6.9 - Steam Generator Low Flow Bypass

10. Section 6.10 - High Logarithmic Power Level

11. Section 6.11 - High LPD and Low DNBR Bistables

IV. SO23-3-3.25, Revision 7

A. Title. Once-a-Shift Surveillance (modes 1-4).

B. Description. Those readings, channel checks, and other surveillances required to be performed once a shift on a routine basis are peformed, including the channel check of the safety channel and PPS.

C. Responsible Group. Operations.

D. Outline

Section 6.4 - Reactor Protective/Engineered Safety Feature Actuation System Instrumentation Channel Checks.

V. SO23-3-3.2, Revision 4

A. Title. Excore Nuclear Instrumentation Calibration.

B. Description. This test determines core power by secondary calorimetric and then adjusts the safety channels to agree with the secondary calorimetric value and with each other.

C. Responsible Group. Operations.

D. Outline

1. Section 6.1 - Power Determination
2. Section 6.2 - Safety Channel Calibration
3. Section 6.3 - Control Channel Calibration

VI. SO23-V-1.19.1, Revision 0

A. Title. Excore Log Power Calibration.

B. Description. The results of this surveillance modify the factory alignment voltages specified in both the 31-day and 18-month instrumentation and control procedures. The information to update the instrumentation and control 31-day surveillance procedures is explicitly provided to instrumentation and control via this procedure. No modification of the 18-month calibration procedure is initiated. That calibration always restores excore alignment to factory specifications.

C. Responsible Group. Station Technical (with instrumentation and control assistance).

D. Outline

1. Section 6.1 - Data Collection

2. Section 6.2 - Safety Channel Excore Logarithmic Power Calibration

3. Section 6.3 - Startup Channel Excore Logarithmic Power Calibration

4. Section 6.4 - Restoration

APPENDIX C

VERIFICATION OF SUBSYSTEM FUNCTIONS BY
CURRENT SURVEILLANCE TESTS

TABLE C-1. VERIFICATION OF SUBSYSTEM FUNCTIONS BY CURRENT SURVEILLANCE TESTS

Subsection	Channel Check	Channel Calibration	Channel Functional Test
6.2.1 Power Supply	Not directly checked. SO23-3-3.25 (OPS), paragraph 6.4.1. Verify individual nuclear instrumentation drawer switches in proper position.	1. SO23-II-5.5-5.8 (I&C), paragraph 6.2.1, (31-day). Verifies proper power supply voltages. <u>Range and Accuracy</u> Using DVM +15V ± 0.2V -15V ± 0.2V -800V ± 25V <u>Known Signal</u> Drawer itself measured with DVM. 2. SO23-II-5.1-5.4 (I&C), paragraph 6.6 and 6.7 (18 months). Verifies proper power supply voltages (same as above) and also verifies the bistable setpoint for low voltage on the 800V power supply. <u>Range and Accuracy</u> Same as above. <u>Known Signal</u> Same as above.	Not directly checked.

TABLE C-1 (continued)

Subsection	Channel Check	Channel Calibration	Channel Functional Test
6.2.2 Log Circuits	SO23-3-3.25 (OPS), paragraph 6.4.1. Record the log power readings (four channels and verify all readings within 1/3 decade.	1. SO23-II-5.5-5.8 (I&C), paragraph 6.2.2 (31-day). Using DVM and log, calibrate positions 1 through 6 - verify each output in voltage and meter reading to be within the required range and accuracy. Range and Accuracy: Source SO23-V-1.19.1, paragraph 6.2 and Attachment 4. Known Signal: Voltage output measured by DVM. 2. SO23-II-5.1-5.4 (I&C), paragraph 6.9 through 6.14 (18-month). a. Verifies the wave forms for each of the six positions of the log calibrate selector. b. Verifies the log count rate discriminator threshold using the Tennelec Pulser.	SO23-II-1.1.1 through 1.1.4 (I&C), paragraph 6.10. 1. Turn off excore drawer and separately deenergize H.V. in excore drawer Verify annunciator (56B05,15,25, and and (35) "NI INOPERATIVE CH__." 2. Using log trip test, potentiometer in excore drawer - verify bistable in PPS and control room annunciator 56A12 (pretrip) and 56A02 (trip) setpoints (0.89%) are correctly set.

TABLE C-1 (continued)

Subsection	Channel Check	Channel Calibration	Channel Functional Test
		c. Adjusts the voltage output for each of the six positions of the log calibration switch. d. Performs an alignment check of the indications and voltages for the log power channel. Range and Accuracy: Per this procedure. Known Signal: Tennelec Pulser or voltage output measured by DPM.	
6.2.3 10^{-4} Bistable	1. Not directly checked. S023-3-3.25 (OPS), paragraph 6.4.1. Verify switches in nuclear instrumentation drawer in proper position.	1. S023-II-5.5 through 5.8 (I&C) paragraph 6.2.3. Using DVM and safety drawer "log trip test potentiometer" - verify setpoint and accuracy of 10^{-4} bistable voltage output from nuclear instrumentation drawer.	S023-II-1.1.1 through 1.1.4 (I&C) paragraph 6.8 1. Using the excore safety channel drawer "log calibrate switch," verify that: a. Excore drawer "10^{-4} bistable light" functions properly.

TABLE C-1 (continued)

Sheet 4 of 9

Subsection	Channel Check	Channel Calibration	Channel Functional Test
	2. SO23-3-3.25 (OPS), paragraph C.12. Verify CPC remote operations module 10^{-4}% bypass switch in proper position.	<u>Range Accuracy</u> Specified in step. <u>Known Signal</u> Drawer voltage output measured by DVM. 2. SO23-II-5.1 through 5.4 (I&C), paragraph 6.2.1 (18-month). Adjust the 10^{-4} bistable to trip within the required voltage value. <u>Range and Accuracy</u> Specified in this step. <u>Known Signal</u> Drawer voltage output measured by DVM.	b. ROM "High Log Bypass Off" functions properly. 3. ROM "High Log Power Bypass" light functions properly. 4. Control Room annunciator 56A47, "high log power permissive" operates properly.
6.2.4 Rate Channel	Not directly checked. SO23-3-3.25 (OPS), paragraph 6.4.1. Verify individual nuclear instrumentation drawer switches in proper position.	1. SO23-II-5.5 through 5.8 (I&C), paragraph 6.2.4. Using DVM and "rate calibrate switch," verify ODPM, 7DPM, and alarm setpoint, all within required tolerance.	SO23-II-5.5 through 5.8 (I&C), paragraph 6.2.4. 1. Alarm setpoint and control room annunicator functionally tested by same procedure.

TABLE C-1 (continued)

Subsection	Channel Check	Channel Calibration	Channel Functional Test
		Range and Accuracy — Specified in this procedure. Known Signal — Voltage output measured by DVM. 2. S023-II-5.1 through 5.4 (I&C), paragraph 6.15 (18-month). Adjusts the rate meter circuit for 0 and 7 DPM, corresponding to 0 and 10 volts. Range and Accuracy — Specified in this procedure. Known Signal — Drawer voltage output measured by DPM.	2. PPS 31-day test has no steps to test this - none required.
6.2.5 through 6.2.9 Linear Channel Zero, Gain, and Summer and Output Amp.	1. S023-3-3.25 (OPS) paragraph 6.4.1. Compare all four linear safety channel indicators - must agree within 2% of secondary calorimetric power and CPC indicated power.	1. S023-3-3.2 (OPS), paragraph 6.2.3. Using plant computer-generated secondary calorimetric (CV9005) value and DVM measurement of actual nuclear instrumentation output voltages - adjust all four nuclear instrumentation output voltages to agree with calculated voltage generated from calorimetric. This also adjusts CPC constants to be the same as calorimetric value by calculation.	S023-II-1.1.1 thru 1.1.4 (I&C), paragraph 6.6. 1. Using the linear trip test potentiometer in the excore safety channel drawer, verify functional operation and calibration setpoint of

TABLE C-1 (continued)

Sheet 6 of 9

Subsection	Channel Check	Channel Calibration	Channel Functional Test
	2. S023-3-3.25 (OPS) Attachment 3, item 32, verifies all switch positions and indicating lights in proper position/ indication.	<u>Range and Accuracy</u> Specified in this procedure. <u>Known Signal</u> Secondary calorimetric calculated power (PMS PT.ID. CV9005). 2. S023-II-5.5 through 5.8 (I&C), paragraph 6.2.5 through 6.2.9. a. Using known milliampere input to each linear amplifier, verify 0 and 10 volt calibration of voltage output and meter reading for each amplifier and summed output. b. Verifies using DVM that ROM linear calibrate potentiometer, is calibrated to 10V output to nuclear instrumentation drawer. <u>Range and Accuracy</u> In procedure and from S023-V-1.6. <u>Known Signal</u> Standard milliampere input from calibrated source.	pretrip and trip setpoint by observing both indicated power and output voltage. 2. Also verify ROM indicator lights and annunciator windows 56A11 and 56A01 operate properly.

TABLE C-1 (continued)

Subsection	Channel Check	Channel Calibration	Channel Functional Test
		3. SO23-II-5.1 through 5.4 (I&C), paragraph 6.16-6.20 (18-month). a. Using the calculated current values excore safety channel technical manual group, calibrate linear subchannel gains for each amplifier for the zero, 100% and 200% values. b. Adjust the linear calibrate switch output for zero and 200% to correspond to the current values from the technical manual. c. Verify calibration of the summing circuit. d. Verifies the proper operation of the isolation buffer circuitry. <u>Range and Accuracy</u> Specified in this procedure. <u>Known Signal</u> Voltage output measured by DVM.	

TABLE C-1 (continued)

Subsection	Channel Check	Channel Calibration	Channel Functional Test
6.2.10 55% Bistable	SO23-3-3.25 (OPS), paragraph 6.4.1 and Attachment 3. Verify that above 55% power, the loss of load trip is enabled by the presence of the 55% light on the PPS cabinet.	1. SO23-II-5.5 through 5.8 (I&C), paragraph 6.10. Using the linear trip test control, verify that the 55% bistable trips are within the required tolerance and the light is eliminated. Range and Accuracy: From SO23-V-1.19.1 Known Signal: Actual channel signal and voltage output as measured by DVM.	1. SO23-II-1.1.1 through 1.1.4 (I&C), paragraph 6.7. Verifies that the loss of load trip can initiate when the loss of load bypass annunciator 56A30(40, 50, and 60) is extinguished. Also verifies operability of the loss of load annunciator, using the linear trip test potentiometer in nuclear instrumentation drawer.

Subsection	Channel Check	Channel Calibration	Channel Functional Test
		2. S023-II-5.1 through 5.4 (I&C), paragraph 6.22 (18-month). Calibrate the voltage output corresponding to 55% power and verify that the bistable trips within the required tolerance. Range and Accuracy: Per this procedure. Known Signal: Actual channel voltage output as measured by DVM.	2. S023-II-5.5 through 5.8 (I&C), paragraph 6.2.10. Functionally verifies 55% bistable output and control room annunciator 56A30 (40, 50, and 60) using linear trip test potentiometer in nuclear instrumentation drawer.
6.2.11 CPC Reset	N/A*	N/A NOTE: Performed by Procedure S023-II-5.5 through 5.8 to realign equipment to "operable" status after performance of this test.	N/A
6.2.12 Steam Generator Low Flow Bypass Reset	N/A	N/A NOTE: Performed by Procedure S023-II-5.5 through 5.8 to realign equipment to "operable" status after performance of this test.	N/A

*N/A = not applicable.

APPENDIX D

NUCLEAR INSTRUMENTATION SURVEILLANCE IMPLEMENTATION OF TECHNICAL SPECIFICATIONS AT OTHER UTILITIES

A number of other utilities have the same General Atomic safety channels as those at SONGS Units 2 and 3. These include:

Utility	Nuclear Plant
Arkansas Power & Light Company	ANO-2
Louisianna Power & Light Company	Waterford 3
Boston Edison Company	Pilgrim Station
Arizona Public Service Company	Palo Verde 1, 2, and 3

A comparison between SONGS and other utilities with the same excore safety channels has potential benefits because each utility may assign different groups (i.e., operations or instrumentation and control) and have a different procedural organization to satisfy the same technical specification requirements. The comparison could yield cases for which the utility has increased system availability and reduced manpower requirements by simply reorganizing the procedures into a more logical and effective format.

The Arkansas Power & Light Company and Arizona Public Service Company provided information of surveillance testing policies for the nuclear instrumentation safety channel drawers.

Table D-1 provides specific information on how each of these utility's surveillances on the nuclear instrumentation safety channels compares with the methods presently used by Southern California Edison Company. Figures D-1 and D-2 provide reproduced copies of the actual technical specifications for ANO-2 and Palo Verde 1, respectively (References D-1 and D-2).

Both plants have technical specifications that are very similar to those for SONGS Units 2 and 3. The major differences in the method of surveillances are

- Palo Verde
  - Power supplies are checked on an 18-month basis only. (SONGS is checked monthly.)

- To satisfy the linear subchannel gain requirement, the "linear calibrate" switch is used in a monthly test, and a milliampere source is used on the quarterly test (similar to SONGS monthly test.)

- ANO-2

  The prestartup requirement for log channel functional test is completed by the operations group as part of the operations startup procedure. This requirement at SONGS is met by the Instrumentation and Control Department.

REFERENCES

D-1. Arkansas Nuclear One - Unit 2 Technical Specifications Appendix A to License No. NPF-6.

D-2. Technical Specifications, Palo Verde Nuclear Generating Station, Unit 1, Docket No. 5-528, Appendix A to License No. NPF-41.

TABLE D-1. COMPARISON OF TECHNICAL SPECIFICATION SURVEILLANCES ON NUCLEAR INSTRUMENTATION SAFETY CHANNELS

Sheet 1 of 2

Channel	SONGS Requirement	Comparison with Palo Verde	Comparison with ANO-2
Log-Channel	Shift - Channel Check (operations)	Same as SONGS.	Same as SONGS.
	Monthly Functional Test (instrumentation and control)	Same as SONGS.	Same as SONGS.
	Startup Functional Test (instrumentation and control)	Same as SONGS.	Performed by Operations Department as part of startup procedure.
	Refueling Channel Calibration (instrumentation and control)	Same as SONGS.	Same as SONGS except does not use Tennelec Pulser.
Linear Channel	Shift - Channel Check (operations)	Same as SONGS.	Same as SONGS.
	Daily Channel Calibration	Same as SONGS.	Same as SONGS.

TABLE D-1 (continued)

Sheet 2 of 2

Channel	SONGS Requirement	Comparison with Palo Verde	Comparison with ANO-2
Linear Channel (continued)	Monthly Channel Calibration (instrumentation and control)	Same as SONGS except: • Power supplies not checked. • Linear subchannel gains verified using "linear calibrate" potentiometer, as opposed to using a milliampere source.	Same as SONGS.
	Quarterly Channel Calibration (instrumentation and control)	Same as SONGS except different procedure used for quarterly versus monthly tests. Quarterly procedure includes use of milliampere source, as at SONGS.	Same as SONGS.
	Refueling Channel Calibration (instrumentation and control)	Same as SONGS.	Same as SONGS.
	Monthly Functional Test (instrumentation and control)	Same as SONGS.	Same as SONGS.

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FIGURE D-1. ARKANSAS UNIT 2 TECHNICAL SPECIFICATION
TABLE 4.3-1, REACTOR PROTECTION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE 4.3-1

REACTOR PROTECTION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TESTS	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. Linear Power Level - High	S	D(2,4), M(3,4), Q(4)	M	1, 2
3. Logarithmic Power Level - High	S	R(4)	M and S/U (1)	1, 2, 3, 4, 5 and *
4. Pressurizer Pressure - High	S	R	M	1, 2
5. Pressurizer Pressure - Low	S	R	M	1, 2 and *
6. Containment Pressure - High	S	R	M	1, 2
7. Steam Generator Pressure - Low	S	R	M	1, 2 and *
8. Steam Generator Level - Low	S	R	M	1, 2
9. Local Power Density - High	S	D(2,4), R(4,5)	M, R(6)	1, 2
10. DNBR - Low	S	S(7), D(2,4), M(8), R(4,5)	M, R(6),	1, 2
11. Steam Generator Level - High	S	R	M	1, 2
12. Reactor Protection System Logic	N.A.	N.A.	M	1, 2 and *
13. Reactor Trip Breakers	N.A.	N.A.	M	1, 2 and *
14. Core Protection Calculators	S, W(9)	D(2,4) R(4,5)	M, R(6),	1, 2
15. CEA Calculators	S	R	M, R(6),	1, 2

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

* - With reactor trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

(1) - If not performed in previous 7 days.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust the Linear Power Level signals and the CPC addressable constant multiplers to make the CPC ΔT power and CPC nuclear power calculations agree with the calorimetric calculation if absolute difference is >2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation (conservatively compensate for measurement uncertainties) or by calorimetric calculations (conservatively compensated for measurement uncertainties) and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations (conservatively compensated for measurement uncertainties).

(9) - The correct values of addressable constants shall be verified to be installed in each OPERABLE CPC.

FIGURE D-2. PALO VERDE UNIT 1 TECHNICAL SPECIFICATIONS
TABLE 4.3-1, REACTOR PROTECTIVE INSTRUMENTATION
SURVEILLANCE REQUIREMENTS