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REACTOR PROTECTION SYSTEM
SURVEILLANCE TEST EXTENSIONS

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I. Executive Summary

The FitzPatrick plant will be operating on a 24 month fuel cycle. This longer cycle length has a direct effect on surveillance, maintenance, and test activities that are currently performed on a 18 month or refuel outage basis.

At FitzPatrick, the Reactor Protection System (RPS) scram instrument circuits are routinely inspected, tested, and maintained to provide a highly reliable protection system. Typically, the instrument circuits are subjected to a channel check each shift, a monthly functional test, and an 18 .nonth channel calibration, alignment, and, when required, a response time test.

RPS test frequencies are mandated by the plants' Technical Specifications. Maintenance activities are based on operational feedback and manufacturer's recommendations.

This study evaluates changes to surveillance requirements to support a nominal twenty four month fuel cycle. Justification is provided, where appropriate, to support test and maintenance interval extensions.

Our evaluations conclude that 1) the RPS 18 month channel calibrations, alignments, and response time tests can be safely extended to support a nominal 24 month operating cycle, and 2) Technical Specification changes are required to extend some of the surveillance tests.

24 MONTH OPERATING CYCLE
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II. Purpose

The FitzPatrick plant will be operating on a 24 month fuel cycle. To avoid either an 18 - month surveillance outage or an extended mid-cycle outage, changes are required to the RPS surveillance test intervals prescribed by the FitzPatrick Technical Specifications.

Substantiating the effects of the longer cycle length on surveillance, maintenance, and test activities requires a comprehensive review of the RPS instrument system, its individual components, and the integrated effect of test extensions on system operability. Consequently, this report also contains recommended improvements in RPS surveillance tests which are not directly impacted by the longer fuel cycle.

III. Evaluation

RPS test, maintenance, and inspection activities were methodically evaluated to determine the impacts of a 24 month operating cycle. The longer cycle length requires an extension of 18 month surveillance test intervals for RPS channel calibrations, alignments and response time testing. Other RPS surveillance tests, such as, channel checks and channel functional tests are performed during power operation and are not directly impacted by the longer fuel cycle. Similarly, RPS preventive maintenance activities are either performed during power operation or are not cycle length dependent. The longer cycle length requires an evaluation of the following "refueling interval" calibrations and surveillance tests:

- Turbine stop valve closure (RPS) instrument calibration (ISP-37)
- APRM flow bias signal instrument calibration (ISP-63-1)
- Scram discharge instrument volume high water level instrument functional test/calibration (ISP-66-1)
- Scram discharge instrument volume water level transmitter calibration (ISP-66-4)
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- Mode switch in shutdown functional test (ST-29B)
- RPS channel test switch functional test (ST-29C)
- Backup scram valves functional test (ST-29E)

The decision to extend surveillance test intervals considers:

1. the protection system's safety function; the logic circuitry, instrument loops, accuracy and drift of individual components which accomplish this function, and

2. the integrated effect of testing and maintenance activities on system operability.

These considerations are applied to the FitzPatrick Reactor Protection System:

1. <u>System Safety Function</u>

The Reactor Protection System (RPS) provides protection against the onset and consequences of conditions that threaten the integrity of the fuel cladding and the reactor coolant pressure boundary. The RPS limits the uncontrolled release of radioactive material from the fuel and the reactor coolant pressure boundary by terminating excessive temperature and pressure increases through the initiation of an automatic scram.

The Reactor Protection System is comprised of two independent Trip Systems (A and B) each powered from an AC motor generator set. Each trip system has two auto-scram trip channels (trip channels A1 and A2, B1 and B2) and one manual trip channel (A3 and B3). The outputs of the auto-scram trip channels are combined in a logic so either channel will trip the associated Trip System. The tripping of both trip systems will produce a reactor scram.

There are two scram pilot valves for each control rod drive Hydraulic Control Unit (HCU). Each scram pilot valve is solenoid operated, with

the solenoids normally energized. The scram pilot valves control the air supply to the scram inlet and outlet valves for their associated HCU. When either scram pilot valve solenoid is energized, air pressure holds the scram valves closed and therefore, both scram pilot valve solenoids must be de-energized to cause a control rod to scram. The scram valves control the supply and discharge paths for the control rod drive water during a scram. One of the scram pilot valve solenoids for each HCU is controlled by Trip System A, and the other solenoid is controlled by Trip System B. Any trip of Trip System A in conjunction with Trip System B results in both solenoids de-energized, air bleed off, scram valves opening and control rod scram. The backup scram valves (which energize on a scram signal to depressurize the scram air header) are also controlled by the RPS.

The RPS includes sensors, relays, bypass circuits, and switches that are necessary to cause initiation of a reactor scram. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. The plant conditions which are monitored by the reactor protection system and used to cause a scram are:

- High neutron flux
- Reactor coolant system high pressure (High reactor pressure)
- Reactor vessel low water level
- Turbine stop valve closure
- Turbine control valve fast closure
- Main steam line isolation
- Scram discharge volume high water level
- Primary containment high pressure (High drywell pressure)
- Main steam line high radiation
- Manual scram
- Reactor mode switch in shutdown

The safety objective of the Neutron Monitoring System is to detect conditions in the core that threaten the overall integrity of the fuel barrier due to excessive power generation, and to signal the Reactor

Protection System, so as to prevent the release of radioactive materials from the fuel barrier. Specific portions of the Neutron Monitoring System that are required for safety are the Intermediate Range Monitor (IRM) subsystem and the Average Power Range Monitor (APRM) subsystem. The IRM subsystem provides a signal to the Reactor Protection System to shutdown the reactor when an abnormal transient occurs in the intermediate range during startup. The APRM subsystem provides a signal to the RPS to shutdown the reactor when an excessive power generation condition exists in the power range. For reactor operation in the startup mode, the APRM subsystem also provides a shutdown signal if an abnormal power transient exceeds 15 percent of rated power.

High pressure within the Reactor Coolant System poses a direct threat of rupture to the Reactor Coolant Pressure Boundary. A RCS pressure increase while the reactor is operating compresses the steam voids and results in a positive reactivity insertion causing increased core heat generation that could lead to fuel failure and system over-pressurization. A scram counteracts a pressure increase by quickly reducing the core fission heat generation.

Low water level in the reactor vessel indicates that the core is in danger of being inadequately cooled. The effect of a decreasing water level while the reactor is operating at power is to decrease the reactor coolant inlet subcooling. Should water level decrease too far, fuel damage could result as steam voids form around fuel rods. A reactor scram protects the fuel by quickly reducing the fission heat generation within the core.

Closure of the turbine stop valve with the reactor at power can result in a significant addition of positive reactivity to the core as the Reactor Coolant System pressure rise collapses steam voids. The turbine stop valve closure scram, initiates a scram earlier than either the Neutron Monitoring System or Reactor Coolant System high pressure. The scram counteracts the addition of positive reactivity due to pressure by quickly inserting negative reactivity with the control rods.

With the reactor and turbine-generator at power, fast closure of the turbine control valves can result in a significant addition of positive reactivity to the core as RCS pressure rises. The turbine control valve fast closure scram is required to provide a satisfactory margin to core thermal hydraulic limits for this category of abnormal operational transients. The scram counteracts the addition of positive reactivity due to pressure by inserting negative reactivity with the control rods. Although the RCS high pressure scram, in conjunction with the Pressure Relief system, precludes overpressurizing the Reactor Coolant system, the turbine valve fast closure scram provides additional margin

to the RCS pressure limit.

The main steam line isolation valve closure scram limits the release of fission products from the Reactor Coolant System. Automatic closure of the main steam line isolation valves is initiated upon conditions indicative of a steam line break. (Low reactor vessel water level, high steam line flow, high steam line tunnel temperature, low steam pressure at the main turbine inlet.) The scram initiated by main steam line isolation valve closure anticipates a reactor vessel low water level scram.

The scram discharge volume receives the water displaced by the control rod pistons during a scram. Should the scram discharge volume fill up with water to the point where not enough space remains for the water displaced during a scram, control rod movement would be hindered in the event a scram were required. To prevent this situation the reactor is scrammed when the water level in the discharge volume attains a value high enough to verify that the volume is filling up, yet low enough to ensure that the remaining capacity in the volume can accommodate a scram.

High pressure inside the primary containment (high drywell pressure) could indicate a break in the Reactor Coolant Pressure Boundary. The reactor is scrammed in such a situation to minimize the possibility of fuel damage and to reduce the addition of energy from the core to the coolant.

High radiation in the vicinity of the main steam lines could indicate a gross fuel failure in the core. When high radiation is detected near the steam lines, a scram is initiated to limit the fission products released from the Reactor Coolant Pressure Boundary. This same high radiation condition also signals the Primary Containment and Reactor Vessel Isolation Control System to initiate containment of the released fission products.

To provide the control room operator with means to rapidly shut down the reactor, two pushbuttons are located in the Control Room which initiate a scram when both are actuated by the control room operator.

The reactor mode switch provides appropriate protective functions for the condition in which the reactor is to be operated. The reactor is SHUTDOWN with all control rods inserted when the mode switch is in the SHUTDOWN position. Placing the mode switch in the SHUTDOWN position initiates a reactor scram. This scram is not considered a protective function because it is not required to protect the fuel or Reactor Coolant Pressure Boundary, and it bears no relationship to minimizing the release of radioactive material from any

barrier.

2. Testing and Maintenance Activities

A review of RPS surveillance testing and maintenance activities was conducted to determine the impact, if any, certain test extensions would have on system reliability and operability. A comparison between on-line and refueling interval surveillance tests was performed for the RPS. The on-line surveillances were reviewed to determine the extent this data might overlap the surveillances performed during refueling.

IV. Calibration Extensions

The FitzPatrick Technical Specifications require the following surveillance tests on Reactor Protection System scram circuitry:

1. a daily channel check; typically, an observation that each channel of an RPS scram variable indicates the same process value. Some RPS channels (manual scram, MSIV closure, SDIV high level float switch, etc.) are not analog instruments and thus are excluded from the daily check requirement.

2. weekly, monthly, or quarterly channel functional tests; verifies proper channel response (trip and alarm functions) to a simulated trip signal. Performing a channel functional test also validates channel-failure alarm circuitry and the trip setpoint.

3. instrument calibrations; typically, once an operating cycle or every refueling outage. Some equipment is calibrated more frequently (e.g., master/slave trip units - 6 months). Calibrations encompass the entire instrument channel. Components are adjusted, when possible, at five points across the instrument span.

4. response time tests; the time interval from when the monitored parameter exceeds its trip setting until the scram pilot valve solenoids are de-energized. RPS response time is measured once an operating cycle.

In addition, functional testing of the RPS channel test switch, mode switch, and backup scram valves is performed at refueling intervals. These tests demonstrate the operability of the RPS automatic and manual scram trip channels and ensure the backup scram system components are functioning properly. A summary listing of each RPS scram circuit and the associated surveillance tests is shown on Table 1.

- Instrument Drift Analysis

The extension of RPS calibration intervals includes a plant specific instrument drift analysis (Reference 43) for each instrument loop. Field calibration data for instruments currently calibrated once every 18 (plus 25% for a maximum of 22.5) months was evaluated to assess the acceptability of extending calibration intervals to 24 (plus 25% for a maximum of 30) months. In general, Instrument Drift Evaluations (IDEs) are comprised of two phases. Phase 1 is an evaluation of past instrument performance compared to theoretical acceptance limits (i.e., vendor drift allowance, VDA or the existing field calibration tolerance, CT). Phase 2 predicts future drift by statistically extrapolating the component's derived drift data to arrive at a value for maximum expected drift over a 30 month interval (MED30).

Past performance is indicated by instrument "drift," which is derived from field calibration data by taking the difference between the "as-found" and "as-left" calibration values. The derived value actually encompasses instrument accuracy, measuring and test equipment uncertainties, and the effects of ambient environmental conditions (temperature, pressure, humidity and radiation) in addition to instrument drift. Therefore the term "drift" that is used throughout this evaluation and in the drift analysis is a misnomer and actually represents total instrument calibration uncertainties.

Generally, if Phase 1 of the IDE shows that a component's derived drift data falls within the vendor drift allowance or the calibration tolerance (with the exception of rare occurrences), past performance is considered acceptable. Deviations are explained on a case-by-case basis. Phase 2 predicts future instrument performance over a maximum 30 month period (MED30) using data obtained from Phase 1. The MED30 value bounds hardware performance with a 95% probability at a 95% confidence level (i.e. there is a 95% probability that 95% of all past, present and future calibration results will be less than the maximum expected drift). Loop accuracy/setpoint calculations are then updated to include 30 month calibration uncertainties.

- Loop Accuracy/Setpoint Calculations

Loop accuracy calculations establish total channel uncertainties by accounting for instrument inaccuracies consistent with industry methods described in ISA RP67.04 (Reference 44). The loop accuracy calculation for an instrument channel uses conservative values for 30 month calibration uncertainties (vendor specified uncertainties or MED30, whichever is larger). If MED30 is overly conservative, vendor specified uncertainty values are used in the calculation. The value calculated for MED30 is considered unrealistic or overly conservative if

too few data points are available, resulting in a high statistical multiplier. The loop accuracy/setpoint calculations must show that sufficient margin exists between the analytical limit and the existing field trip setting in order to be consistent with the assumptions of the safety analysis.

The following RPS instrument channel calibrations were evaluated to determine whether the calibration interval could be safely extended to support a 24 month operating cycle:

1) Turbine Stop Valve Closure (RPS) Instrument Calibration (ISP-37)

The turbine stop valve closure scram prevents the addition of positive reactivity to the core resulting from closure of the turbine stop valves. This scram counteracts the addition of positive reactivity due to pressure by inserting negative reactivity with the control rods. Turbine stop valve closure inputs to the Reactor Protection System are from valve stem position switches mounted on the four turbine stop valves. Each of the double-pole, single throw switch is arranged to open before the valve is more than 10% closed to provide the earliest positive indication of closure. Either of the two channels associated with one stop valve can signal valve closure. The logic is arranged so that closure of three or more valves initiates a scram.

ISP-37 (Reference 16) checks the operability (and calibrates if necessary) the following turbine stop valve (TSV) closure limit switches:

- 94PNS-101, 102
- 94PNS-103, 104

An assessment of past instrument performance was performed to determine whether individual drift values for a given calibration interval were within acceptable limits (ISP-37 specifies a calibration tolerance for the valve position switches of 1% of valve closure). Calibration data from April 1987, November 1988, and June 1990 was evaluated by the instrument drift analysis. 1992 calibration results were excluded from this review since the limit switches (94PNS-101 through 104) were replaced during the 1992 outage. Actual past instrument drift was established by comparing "as-found" channel conditions to the previous calibration "as-left" conditions. This review confirmed that past drift values for the limit switches were not time dependent (as documented in Attachment 3) and were within the specified calibration tolerance except on a rare occasion.

Field calibration data was also used to conservatively predict future instrument drift. The maximum expected drift for 30 months (MED30)

was determined from the average and standard deviation of calibration data statistically extrapolated to 30 months. The MED30 value for the limit switches is approximately 1.9% of valve closure. This value has a 95% probability of occurrence at a 95% confidence level.

The existing trip setpoint for the turbine stop valve limit switches is 8% (± 1%) of valve closure. This trip setting provides a 2% margin between the trip point and the Technical Specification limit to account for instrument loop inaccuracies. Applying 95/95 projections of instrument drift (MED30) to the instrument loop reduces this margin to 0.1%. In order to avoid potential Technical Specification violations, it is recommended that the field trip setting be changed from 8% to 7% of valve closure (see Table 2). This change will provide additional margin for channel uncertainties without interfering with routine operation of the plant (Reference 45). NOTE: NEDC-31336 recommends a nominal trip setpoint of 94% of full open valve position (6% closed) for the Turbine Stop Valve Closure scram setting. The recommended setpoint change is within the trip setting limits determined by the GE analysis.

In addition to the "refueling interval" calibration, the Technical Specifications also require a monthly functional test (Reference 7) of the main turbine stop valves position switches. In this test, two TSVs are closed simultaneously to the 90% open position in order to actuate the RPS limit switches. The channel relays are verified to de-energize and alarms are checked for proper operation. As a result, on-line testing verifies that the relays and alarms are functioning as required when the turbine stop valves reach the 10% closed position.

The turbine stop valve closure instrument calibration can be safely extended with the longer operating cycle because: 1) a review of past instrument performance shows that actual instrument drift has been within the required tolerance band except on a rare occasion, 2) limit switch performance is not time dependent, 3) conservative extrapolations to 95/95 will not exceed Technical Specification limits, and 4) on-line testing provides assurance that the turbine stop valve closure limit switches and their associated circuitry are functioning as required. Furthermore, a trip setting change from 8% to 7% closure will provide further assurances that the longer calibration interval will not result in Technical Specification violations.

2) APRM Flow Bias Signal Instrument Calibration (ISP-63-1)

The Average Power Range Monitor (APRM) subsystem of the Neutron Monitoring System provides a signal to the Reactor Protection System to shutdown the reactor when an excessive power generation condition exists in the power range.

For the APRM flow bias circuit, the APRMs receive flow signals from the Reactor Water Recirculation (RWR) flow unit subsystem. Each flow channel consists of two RWR flow transmitters, a flow unit, remote and local indicators, switches, and remote recorders. Loop A and loop B RWR system driving transmitters provide inputs to the flow units in each flow channel. The flow unit is comprised of 2 square root converters and a summer circuit. The square root converters transform the 10-50 mA current signal from each flow transmitter into a 1 to 10 volt signal that is proportional to the RWR flow. The summer circuit sums the two flow signals and outputs them to the flow control reference circuit in each Rod Block Monitor and APRM. This signal is sent to the trip circuits.

ISP-63-1 (Reference 17) calibrates the following transmitters, flow units, square root converters, and summers:

07FU-14A, B, C, D	02FT-110E, F, G, H
02FT-110A, B, C, D	02SQX-152A, B, C, D
02SQX-152E, F, G, H	02SUM-153A, B, C, D

Past performance of the APRM flow bias instrumentation was evaluated to determine whether individual drift values for a given calibration interval were within acceptable limits. The results of this review are as follows:

- Past instrument drift for the flow transmitters (manufactured by Barton and Foxboro) routinely exceeded the specified calibration tolerance. These transmitters are presently calibrated on an increased frequency and were replaced during the 1993 mini-outage with Rosemount transmitters under Modification F1-87-099 RWR (Reference 49).

- Drift values for the flow units (square root converters, summers, etc.) also exceeded the calibration tolerance on more than rare occasions, but these results were generally repeatable and predictable and therefore can be calibrated on a longer interval. NOTE: See Table 3 for recommended changes to the calibration tolerance for the summers and the square root convertors.

Field calibration data was also used to predict future instrument drift at 95/95 for the flow units. The maximum expected drift for 30 months (MED30) was determined from the average and standard deviation of actual calibration data statistically extrapolated to 30 months. For the flow units, MED30 was less than the vendor drift allowance for 30 months (as calculated from manufacturers data), indicating that the flow units can be extended to a longer calibration interval with a future

drift value of 0.089 Volts. MED30 was also calculated for the flow transmitters, but these values are meaningless since these transmitters were replaced this year.

To date, two loop accuracy calculations for the APRM flow biasing instrumentation (Reference 46) have been completed to justify extension of the calibration interval to a maximum of 30 months. JAF-CALC-RWR-00662 (Reference 47) calculates the channel uncertainties for the indication, recording and computer point portions of the flow biasing loop; Calculation JAF-CALC-NMS-00758 (Reference 48) determines the channel uncertainties for the flow biased and non-flow biased APRM trip circuitry. Both calculations apply 30 month drift values for the new Rosemount model 1153DF transmitters.

RECOMMENDATION: It is suggested that instrument performance be monitored since the installation of the Rosemount transmitters. Based on the conclusions of Calculation JAF-CALC-NMS-00758, calibration of these transmitters could be extended with the longer interval provided a pattern of repeatable, acceptable drift values is established.

3) Scram Discharge Instrument Volume High Water Level Instrument Functional Test/Calibration (ISP-66-1)

The Control Rod Drive Scram system is designed so that all of the water which is discharged from the reactor by a scram can be accommodated in the discharge piping. During normal operation the discharge volume is empty; however, should it fill with water, the water discharged to the piping from the reactor could not be accommodated, which would result in slow scram times or partial control rod insertion. To preclude this occurrence, level detection instruments have been provided in each instrument volume which alarm and scram the reactor when the volume of water reaches the setting limit. In order to ensure diversity and redundancy in scram discharge instrument volume level measurements, inputs to the Reactor Protection System are from four level switches and four differential pressure transmitters/master trip units. There are two level switches and two DPTs individually connected to each of the two scram discharge instrument volumes.

ISP-66-1 (Reference 19) calibrates and functionally tests the following SDIV level switches:

03LS-231A,B
03LS-231E,F

An assessment of past instrument drift (1988 - 1992) was performed to determine whether individual drift values for a given calibration interval were within acceptable limits (ISP-66-1 specifies a calibration tolerance of 0.5" H_20). Instrument drift for the SDIV level switches was

established by comparing "as-found" channel conditions to the previous calibration "as-left" conditions. This review confirmed that past drift values were within the calibration tolerance except on rare occasions.

Future instrument performance was conservatively predicted for the SDIV level sensors. The maximum expected drift for a 30 month cycle has a 95% probability of occurrence at a 95% confidence level. Although MED30 exceeded the calibration tolerance for a 30 month calibration interval, this value is considered to be realistic. Future drift values for the level switches are expected to be within MED30.

Calculation JAF-CALC-CRD-01272 (Reference 50) was prepared to evaluate the effect of an increased calibration interval on SDIV high water level instrumentation. The Technical Specification trip setting limit (Allowable Value) is 34.5 gallons of water, or 24.5" H_20 and the field trip setpoint is 23.0" H_20. The calculation combined MED30 with other instrument loop uncertainties to obtain a total channel uncertainty of 1.44" H_20. This value is less than the 1.5" H_20 of margin provided between the existing trip setpoint and the Tech Spec limit. As a result, the existing trip setpoint is conservative and no change is required to support a 30 month calibration interval.

The scram discharge instrument volume high water level instrument functional test/calibration can be safely extended with the longer operating cycle because: 1) past instrument drift for the level switches was within the specified calibration tolerance except on rare occasions, 2) future instrument drift was conservatively predicted and applied to the loop accuracy/setpoint calculations, and 3) the setpoint calculation for SDIV high water level shows sufficient margin is available to accommodate postulated uncertainties associated with a 30 month calibration interval. In addition, a monthly functional test of the level switches verifies that the appropriate alarms annunciate and relays de-energize when the trip value is reached. In addition, out-of-calibration conditions are also detected during the functional test.

4) Scram Discharge Instrument Volume (SDIV) Water Level Transmitter Calibration (ISP-66-4)

Scram discharge volume high level inputs to the Reactor Protection System are from eight level instruments (four level switches and four differential pressure transmitter/master trip units). The float sensors and the differential pressure transmitters (DPT) are located in the Reactor Building; the master trip units and relays associated with each of the four DPTs are located in the Relay Room within the Analog Transmitter Trip System (ATTS) cabinets. Two level switches and two DPTs are individually connected to each of the two scram discharge instrument volumes to assure diverse and redundant level measurement.

Each level instrument connected to an SDIV provides an input into one channel. The instruments are arranged in pairs so that no single event prevents a reactor scram due to scram discharge volume high water level.

The Technical Specifications require that the SDIV differential pressure transmitters be subjected to a monthly functional test and a quarterly calibration. For consistency with similar ATTS circuits, it is suggested that SDIV differential pressure transmitters be calibrated during the refueling outage. The following discussion provides justification for extending the existing test interval from once/3 months to a maximum of 30 months.

ISP-66-4 (Reference 20) tests and calibrates the following DPTs:

03DPT-231C	03DPT-231D
03DPT-231G	03DPT-231H

In accordance with NRC guidance presented in Generic Letter 91-04 (Reference 85), an evaluation of actual calibration data for the DPTs was performed to determine whether past performance of the transmitters supports extension of the calibration interval from once/3 months to a maximum of 30 months. Past instrument drift was determined by comparing "as found" calibration data to the previous "as left" conditions. This review confirmed that past instrument drift values at the maximum drift point were consistently within the theoretical uncertainties specified by the vendor for a quarterly calibration interval.

Field calibration data was also used to predict future instrument drift. The maximum expected drift for 30 months (MED30) was determined from the average and standard deviation of actual calibration data statistically extrapolated to 30 months. This MED30 value has a 95% probability of occurrence at a 95% confidence level. The MED30 value for the SDIV water level transmitters does not provide a realistic prediction of future instrument drift. The statistical extrapolation of quarterly calibration data results in the over-magnification of loop uncertainties which do not reflect true transmitter performance. As a result, MED30 is considered invalid and is not applied to the loop accuracy/setpoint calculations. Instead, the Instrument Drift Evaluation (Reference 43) uses vendor specifications (VDA30) as a basis for predicting future drift values.

To ensure effective operation of the Control Rod Drive system, the current Tech Spec limit requires SDIV level $\leq$ 34.5 gallons. Calculations JAF-CALC-CRD-00234, R1 and JAF-CALC-CRD-00307, R1 (References 51 and 52) recommend a field trip setting of $\leq$ 30.10 gallons to account for instrument loop uncertainties and instrument

drift. This setting provides a margin of 4.4 gallons for overall loop inaccuracies. A comparison between the results of the instrument drift evaluation and the loop accuracy calculations shows that VDA30 for the DPTs is bounded by the values presently used in the calculations (Reference 53). Therefore, no revision to the field trip setting (30.10 gallons) is required since the calculations conservatively assume larger projected calibration uncertainties than the calculated value for VDA30. It is recommended however, that the "as left" tolerance band be reduced to the transmitters reference accuracy as noted in Table 3.

On-line testing provides assurances that instrument failures will be promptly detected during operation. The SDIV level is checked and recorded each day (Reference 6) during the daily surveillance and instrument check. The monthly functional test (References 25 and 26) verifies that the MTUs actuate at the required trip point, and confirms annunciation of appropriate alarms. Furthermore, the SDIV master and slave trip units are calibrated at quarterly intervals.

The scram discharge instrument volume water level transmitter calibration can be safely extended with the longer operating cycle because: 1) future drift values for the SDIV level transmitters are expected to fall within the vendor drift allowance for 30 months (VDA30), 2) Calculations JAF-CALC-CRD-00234 and JAF-CALC-CRD-00307 verify that sufficient margin exists to accommodate 30 month drift values for the differential pressure transmitters, and 3) on-line functional testing/calibration of MTUs and daily channel checks provide assurance that the SDIV components are operating as required.

5) Turbine Control Valve Oil Pressure (ISP-69-1)

Turbine control valves fast closure initiates a scram based on pressure switches sensing electro-hydraulic control (EHC) system oil pressure. The switches are located between fast closure solenoids and the disc dump valves, and are set relative ($500 < P < 850$ psig) to the normal EHC oil pressure of 1600 psig so that based on the small system volume, they can rapidly detect valve closure or loss of hydraulic pressure. ISP-69-1 (Reference 21) calibrates the pressure switches 94PS-200A, B, C and D.

Past instrument drift was evaluated for each of the turbine control valve oil pressure switches to determine whether individual drift values for a given calibration interval were within acceptable limits (ISP-69-1 specifies a calibration tolerance of 25 psig). Actual past instrument drift from 1986 through 1991 was evaluated. In December 1991 these switches were replaced with the same make and model number (Barksdale TC-9622-3) after the electrical connections to the switches were found to be loose and the switch housings were worn. Corrective

action was taken to ensure no recurrence of this problem. A review of calibration data shows that past drift values for the pressure switches fell within the calibration tolerance 81.25% of the time.

In addition to assessing past instrument drift, field calibration data was used as a basis to predict future instrument drift. The maximum expected drift for 30 months (MED30) was determined from the average and standard deviation of actual calibration data statistically extrapolated to 30 months. This MED30 value has a 95% probability of occurrence at a 95% confidence level. Based on MED30, future drift for the pressure switches was conservatively predicted to be 56.3 psig.

Generic Letter 91-04 requires that projections of instrument drift for increased calibration intervals be consistent with the values of drift errors used in determining safety system setpoints. Calculation JAF-CALC-MTG-0667 (Reference 54) determined the total channel uncertainty to be 56.3 psig. The Technical Specification required range for the pressure switches is $500<P<850$ psig, with a guide value of 600 psig (575 - 626). The loop accuracy calculation determined a setpoint range of $556.3<P<793.7$. This shows that the current setting (guide range) is conservative, and no setpoint changes are required to accommodate postulated uncertainties associated with the longer operating cycle.

Operability of the main turbine control valve and the instrument channel is validated during the monthly functional test (Reference 10). This test demonstrates that turbine control valve EHC low oil pressure, as sensed between the fast closure solenoid and disc dump valve, will initiate a control valve trip and annunciate the appropriate alarms. As a result, the monthly surveillance provides assurance that the turbine control valve and trip circuitry are functioning as required.

The turbine control valve oil pressure switch calibration can be safely extended with the longer operating cycle because conservative projections of instrument drift were applied to the loop accuracy calculation, and no setting change is required. In addition, the on-line channel functional test verifies control valve and instrument channel operability each month.

6) RPS Electrical Protection Assembly Functional Test/Calibration (ISP-94)

Motor Generator (MG) sets 71RP-1A and B provide 120 VAC power to bus A and bus B. Transformers PT-05-6A and B provide the buses with an alternate supply of power. Panels 05-6A and B are power distribution panels containing fuses, circuit breakers and relays and are

used to connect power from buses A and B to systems in the Neutron Monitoring system and RPS.

The Electrical Protection Assembly (EPA) provides redundant protection to the Reactor Protection System, and other essential circuits against overvoltage, undervoltage, and underfrequency conditions. Each EPA consists of trip components which disconnect circuitry from input power whenever voltage or frequency exceed their normal setting. Two EPA's are connected between each of the RPS MG sets and the buses. Also, two series EPA's are connected between the alternate power source and the RPS bus transfer switch.

The Technical Specifications require a once per operating cycle demonstration of the operability of overvoltage, undervoltage, and underfrequency protective instrumentation through performance of a channel calibration. This includes simulated automatic actuation of the protective relays, tripping logic, and output breakers, as well as verifying the setpoints and time delays. ISP-94 (Reference 22) checks and calibrates the following instruments:

71EPA-RPS1A1G, 1A2G, 1A1T, 1A2T
71EPA-RPS1B1G, 1B2G, 1B1T, 1B2T

Additionally, the Technical Specifications require a channel functional test of the RPS Electrical Protection Assemblies each time the plant is in cold shutdown for a period of more than 24 hours unless performed in the previous six months. This test verifies overvoltage, undervoltage, and underfrequency trip setpoints and performs a trip point calibration if the values are not within the specified tolerance band. The functional test also ensures that EPA circuit breakers and indicators operate as required. As a result, cold shutdown testing confirms operability of the EPA protective functions.

Past instrument drift was evaluated for each of the electrical penetration assembly functions (i.e. overvoltage, undervoltage and underfrequency protection), and their associated time delays by comparing "as found" channel conditions to the previous "as left" calibration conditions. This review confirmed that past instrument performance at the maximum drift point has been consistently within the theoretical uncertainties specified by the vendor.

In addition to assessing past instrument drift data, field calibration data was also used to predict future instrument drift. The maximum expected drift for 30 months (MED30) was determined. This value has a 95% probability of occurrence at a 95% confidence level. With the exception of the undervoltage and underfrequency time delays, the MED30 for the maximum drift point is less than the vendor drift

allowance for 30 months (VDA30), as calculated from manufacturers data.

In accordance with Generic Letter 91-04, projections of instrument drift for 30 month calibration intervals were incorporated into safety system setpoint analyses. The overvoltage, undervoltage and underfrequency setpoints and their associated time delays were evaluated for the normal (MG set) source and the alternate feeder.

- JAF-CALC-ELEC-00757, Rev. 4 and JAF-CALC-ELEC-00761, Rev. 2 (References 55 and 56) calculate instrument setpoints for the Normal and Alternate supply feeders considering hardware drift and uncertainties for a 30 month cycle (24 months ± 25%). These calculations determined the existing trip settings for the overvoltage and underfrequency trips are conservative. Setpoint changes associated with the undervoltage relays were completed during the 1994 maintenance outage. The Technical Specification changes are recommended on Table 2. Calibration tolerance changes are noted on Table 3. R1

- The Technical Specifications trip setting limit for the Normal and Alternate feeder time delays is ≤ 4 seconds. JAF-CALC-ELEC-00783, Rev. 1 and JAF-CALC-ELEC-00784, Rev. 1 (References 57 and 58) calculate undervoltage, overvoltage and underfrequency time delay trip setpoints for the Alternate and Normal supply feeders using 30 month drift values. These calculations conclude that the calibration interval for the EPA time delays can be safely extended with the longer operating cycle provided that changes are made to the existing time delay trip setpoints. These changes ensure that sufficient margin is available to accommodate postulated drift and uncertainties associated with a 30 month calibration interval. The setting changes will also reduce the number of spurious trips as recommended by GE SIL-496, Rev. 1 and improve overall EPA reliability (Reference 59).

The Reactor Protection System Electrical Protection Assembly Calibration can be safely extended with the longer operating cycle because conservative projections of instrument drift were applied to the loop accuracy calculations. Recommended setting changes are provided on Table 2. In addition, the more frequent channel functional test provides assurance that the EPA's are operating as required.

7) RPS and PCIS Analog Trip System Master and Slave Trip Unit Alignment (ISP-101A-D)

The master and slave trip units are part of the Analog Transmitter/Trip Unit system. The master trip unit (MTU) is a plug in wire assembly designed to accept a signal from the remote transmitter or accept the input of a three wire RTD. The trip unit contains the circuitry necessary to drive external relays, and provide the desired switching functions and analog output signals. The slave trip units (STU) are used in conjunction with master trip units when it is desirable to have different setpoints from a common transmitter. The slaves obtain their input from an analog output signal of the master trip unit. Procedures ISP-101A-D (References 27, 28, 29 and 30) align the following master and slave trip units which initiate a reactor scram:

02-3MTU-255A,B,C & D	Reactor High Pressure
02-3MTU-201A,B,C & D	Reactor Vessel Low Level
05MTU-212A,B,C & D	Primary Containment High Pressure
05MTU-214A,B,C & D	Turbine First Stage Pressure
03MTU-331C,D,G & H	SDIV High Water Level
03STU-331I & K	SDIV Rod Block
03STU-331J & L	SDIV Not Drained

The following items are checked/tested/calibrated by these procedures for each Master Trip Unit/Slave Trip Unit:

1) Card Out File
2) Trip Status Switches
3) Reset Differential
4) Gross Failure
5) RTD Input Section
6) Analog Output
7) Auxiliary Output

Alignment of the ATTS master and slave trip unit functions listed above is not required by the Technical Specifications. The actuation portion of the MTUs and STUs which initiate a reactor scram are tested during the monthly channel functional tests and calibrated semi-annually. Procedures ISP-101A-D calibrate those portions of the Master and Slave trip units which are not required for reactor scram functions. As a result, the sections of the MTU or STU calibrated by these procedures are primarily used for alarm/indication purposes and are not relied on to perform a safety function.

An instrument drift evaluation for the Rosemount master and slave trip units was performed. This evaluation included the review of a significant amount of calibration data for the MTU/STU functions.

Instrument drift was determined by comparing "as-found" calibration data to the previous calibration "as-left" conditions. In general, past performance of the master and slave trip units was excellent with very repeatable results, consistently within the calibration tolerance.

Field calibration data was also used to predict future instrument drift for the MTUs and STUs with a 95% probability and a 95% confidence level. The maximum expected drift for 30 months (MED30) was determined from the average and standard deviation of actual calibration data statistically extrapolated to 30 months. This review showed that future instrument drift for the master and slave trip units is expected to remain within the specified calibration tolerances for the duration of the longer operating cycle.

The RPS and PCIS analog trip system master and slave trip unit alignment can be safely extended with the longer calibration interval because: 1) alignment of the master and slave trip units is not required by the Technical Specifications, 2) these procedures calibrate sections of the master and slave trip units which are not used for initiation of a reactor scram, 3) past performance of the MTUs and STUs show highly repeatable results consistently within the calibration tolerance, and 4) future instrument drift (for 30 months) is expected to be within the specified tolerance bands.

8) RPS-PCIS (Channels A1,A2,B1,B2) Pressure Transmitter Calibration (ISP-200A through D)

The purpose of these procedures is to check and calibrate the various pressure transmitters used in the Reactor Protection System and Primary Containment Isolation System (PCIS). The functions of these transmitters include detection of high reactor pressure and high drywell pressure, as well as operation of the turbine first stage pressure permissive.

NRC Generic Letter 91-04 requires, in part, that justification for increased calibration intervals be based on: 1) acceptable values for past instrument drift as determined by as-found and as-left calibration data, and 2) predicted future instrument drift values for a 30 month operating cycle (as determined with a high probability and a high degree of confidence) will not present a safety concern.

High Drywell Pressure

Primary containment pressure is monitored by four pressure sensors (05PT-12A,B,C & D) which are mounted on instrument racks outside the drywell in the reactor building. Each sensor provides an input to one channel. Pipes that terminate in the secondary containment

(reactor building) connect the sensors with the drywell interior. The sensors are grouped in pairs, physically separated, and electrically connected to the reactor protection system so that no single event prevents a scram due to primary containment high pressure. These transmitters are also part of instrument loops which initiate automatic closure of various group B valves on receipt of a high pressure signal to prevent the release of significant amounts of radioactive material from the primary containment. Extension of the calibration interval for the primary containment isolation system (PCIS) will be addressed in the PCIS system report.

An assessment of past instrument drift (1986 - 1992) was performed to determine whether individual drift values for a given calibration interval were within acceptable limits. Instrument drift for the primary containment pressure transmitters was established by comparing "as-found" channel conditions to the previous calibration "as-left" conditions. This review confirmed that past drift values were within the theoretical uncertainties specified by the vendor except on a rare occasion.

Field calibration data was also used to predict future instrument drift. The maximum expected drift for 30 months (MED30) was determined from the average and standard deviation of actual calibration data statistically extrapolated to 30 months. This MED30 value has a 95% probability of occurrence at a 95% confidence level. For the pressure transmitters evaluated by this procedure, MED30 exceeded the vendor drift allowance for 30 months (as calculated from manufacturers data). Based on past performance, MED30 is considered to be realistic and future drift for the high drywell pressure sensors is expected to be within MED30.

Generic Letter 91-04 requires that projections of instrument drift for increased calibration intervals be consistent with the values of drift errors used in determining safety system setpoints. Calculation JAF-CALC-RPS-00230 (Reference 60) determined the total channel uncertainty for each instrument loop to be 0.184 psig. This is less than the existing 0.2 psig margin provided by the high drywell pressure trip setting. As a result sufficient margin exists to accommodate channel uncertainties associated with a 30 month calibration interval.

High Reactor Pressure

Reactor pressure is measured at two physically separated locations. A pipe from each location is routed through the primary containment and terminates in the Reactor Building. Two locally mounted, analog transmitter/trip units monitor the pressure in each pipe. The two pairs of analog transmitter/trip units are physically separated and each unit

provides a high pressure signal to one channel. The units are arranged so that each pair provides an input to trip system A and trip system B. The physical separation and the signal arrangement ensure that no single physical event can prevent a scram due to Reactor Coolant System high pressure.

These procedures calibrate the following pressure transmitters: 02-3PT-55A, 02-3PT-55B, 02-3PT-55C, 02-3PT-55D.

An instrument drift evaluation for the reactor high pressure transmitters was performed by comparing "as-found" and "as-left" data. This review shows that past performance of the transmitters was within vendor specified uncertainties 100% of the time.

Future instrument performance was conservatively predicted for the pressure sensors. The maximum expected drift for a 30 month cycle has a 95% probability of occurrence at a 95% confidence level. Although MED30 exceeded vendor allowances for a 30 month calibration interval, this value is considered to be realistic. Future drift values for the high reactor pressure transmitters are expected to be within MED30.

NOTE: Rosemount pressure transmitter 02-3PT-55A was replaced in April 1992 because this device was considered suspect to loss of oil problems. Since the replacement, the transmitter was found reading incorrectly on several occasions as compared to the other reactor pressure transmitters. Consequently, on March 18, 1993 02-3PT-55A was replaced again.

Setpoint calculations for the reactor high pressure trip (References 61 and 62) were revised to reflect the results of the instrument drift evaluation. MED30 was incorporated into the calculations to determine the effect of an increased calibration interval on overall instrument loop accuracy. The revised calculations verify that sufficient margin exists between the existing high reactor pressure trip setpoint and the analytical limit to accommodate postulated drift and uncertainties associated with a 30 month calibration interval.

Turbine First Stage Pressure Scram Bypass

Four turbine first stage pressure sensors (05PT-14A,B,C and D) are provided to initiate the automatic bypass of the turbine control valve fast closure and turbine stop valve closure scrams when the first stage pressure is below 30% of rated thermal power. These sensors are locally mounted in the turbine building and arranged so that no single failure can prevent a valve closure or turbine control valve fast closure scram.

A review of past calibration data for the turbine first stage pressure transmitters shows that these devices performed within vendor specifications 100% of the time. Future instrument drift was conservatively predicted at 95/95 for the pressure transmitters. The maximum expected drift for 30 months (MED30) bounds all past, present and future data with a 95% probability at a 95% confidence level.

The loop accuracy/setpoint calculation for turbine first stage pressure scram bypass (Reference 63) was evaluated to determine the effect of an increased calibration interval on overall instrument loop accuracy. This review determined that the total calibration uncertainty (based on calibration procedures and vendor performance specifications) bounds the maximum expected drift for 30 months as determined by the instrument drift evaluation. As a result, no revision is required to the existing setpoint calculation since uncertainties associated with the longer calibration interval can be safely accommodated (Reference 64).

In addition to the refueling interval calibrations, the Technical Specifications also require a daily channel check, monthly functional test, and a semi-annual calibration of the master trip units for the high drywell pressure, high reactor pressure, and turbine first stage pressure functions. The channel checks (Reference 6) confirm that the instrument loops are intact and will identify excessive drift in the instrument transmitter. The functional tests (References 23, 24, 25 and 26) verify that the MTUs actuate at the required trip point, confirm annunciation of appropriate alarms, and reveal failures in the logic circuitry. Furthermore, the semi-annual calibration of the master trip units (References 27, 28, 29 and 30) provide additional assurance that problems with the instrument loop will be detected on-line.

9) RPS-PCIS Reactor Level Transmitter Calibration and Channel Functional Test (ISP-201A through D)

Reactor vessel low water level signals are initiated from differential pressure sensors (arranged in pairs) which sense the difference between the pressure due to a reference column of water and the pressure due to the actual water level in the vessel. Two instrument lines attached to taps, one above and one below the water level, on the reactor vessel are required for the differential pressure measurement for each pair of sensors. The two pairs of lines terminate outside the primary containment and inside the Reactor Building; they are physically separated from each other and penetrate the vessel at widely separated points. The physical separation and signal arrangement ensure that no single physical event can prevent a scram due to reactor vessel low water level.

NRC Generic Letter 92-04 and Bulletin 93-03 described NRC concerns regarding the potential for inaccuracies in the reactor vessel level monitoring/indication systems found in BWRs. In these notices, the NRC contends that the accuracy of the reactor vessel level system may be challenged by the presence of non-condensible gases dissolved in the reference leg of the BWR water level instrumentation. These dissolved gases could result in a false high level indication (non-conservatively) during depressurization. Bulletin 93-03 notified holders of BWR operating licenses that they should implement modifications to the Reactor Water Level Indicating system to ensure that the water level system design provides high reliability. To address this potential problem, FitzPatrick Modification F1-93-075 installed a backfill system for all five reactor vessel water level reference legs. This modification improves the existing reactor water level indication reliability by mitigating the diffusion of non-condensible gases down the reference leg by maintaining a continuous backflow up the leg during normal operation. This modification does not affect the accuracy and/or calibration results of the level transmitters, and therefore won't affect the extension of calibration intervals.

The following level transmitters are calibrated by procedures ISP-201A through D (References 35, 36, 37 and 38):

02-3LT-101A
02-3LT-101B
02-3LT-101C
02-3LT-101D

In accordance with NRC Generic Letter 91-04, an assessment of past performance must confirm that instrument drift has been within acceptable limits except on rare occasions. Past instrument drift for the reactor low water level transmitters was determined by comparing "as-found" data to the previous calibration "as-left" conditions. A limited amount of data was available for these components due to: 1) modification of the reactor vessel water level system (Reference 65), 2) replacement of one transmitter considered suspect to loss of fill oil, and 3) changes to the calibration tolerances. However, for the data which was available (a total of nine data points), instrument drift was consistently within vendor specified uncertainties.

Future instrument drift was also predicted using field calibration data. The maximum expected drift for 30 months (MED30) was determined from the average and standard deviation of actual calibration data statistically extrapolated to 30 months. Even though limited data was available, future instrument drift was predicted at 95/95. Due to high statistical multipliers resulting from the scant amount of data, MED30 exceeded the vendor drift allowance for 30 months (VDA30) by 0.44%.

Although future drift values are expected to be within vendor allowables, MED30 was conservatively used in calculating the reactor low level trip setting.

The Technical Specification Reactor low water level scram setting is ≥ 177 inches above the top of the active fuel (TAF). The existing field trip setpoint is 178.2 inches above TAF. This setting provides a margin of 1.2 inches for instrument loop inaccuracies. MED30 and other channel uncertainties were applied to the setpoint/loop accuracy calculations for the Reactor Low Water Level 3 Scram (References 66 and 67). These calculations conservatively determined that the setpoint for Reactor low water level should be raised to 179.6 inches above TAF to accommodate a maximum 30 month calibration interval. Operating procedure 2A indicates that the reactor water level varies between 196.5 and 206.5 inches above TAF during normal operating conditions. Therefore, raising the trip setting to 179.6 inches won't significantly increase the likelihood of spurious trips.

In addition to the refueling interval calibration of the transmitters, on-line testing also verifies the functionality of the reactor water level instrumentation. Daily channel checks confirm that the instrument loops are intact and will detect transmitter failures (Reference 6). Each month the MTUs are functionally tested (Reference 23, 24, 25 and 26) as required by the Technical Specifications. This test reveals failures in the logic circuitry as well as verifying that the MTUs trip at the required values and appropriate alarms annunciate. The master trip units are also calibrated once/6 months (References 27, 28, 29 and 30).

The RPS reactor level transmitter calibration and channel functional test can be safely extended with the longer operating cycle because conservative projections of instrument drift were applied to the loop accuracy/setpoint calculations. Recommended setting changes are provided on Table 2. In addition, on-line surveillance tests provide assurances that the transmitters are functioning properly.

10) Main Steam Line Isolation Valve 10% Closure (RPS) Limit Switch Calibration Surveillance (MST-29.2)

Main steam line isolation valve (MSIV) closure inputs to the Reactor Protection System are from valve stem position switches mounted on the eight main steam line isolation valves. Each of the double-pole, double-throw switches is arranged to open before the valve is more than 10% closed to provide the earliest positive indication of closure. Each trip logic receives signals from the valves associated with two steam lines. The logic/circuitry arrangement ensures that in no case does closure of two valves or isolation of two steam lines cause a scram

due to valve closure. Closure of one valve in three of four of the steam lines causes a scram.

MST-29.2 (Reference 39) calibrates the following MSIV limit switches:

29PNS-80A2, 80B2, 80C2, 80D2
29PNS-86A2, 86B2, 86C2, 86D2

The MSIV limit switches are NAMCO Model No. EA740-80100. These switches send an initiation signal to the reactor protection system before the main steam isolation valve is more than 10% closed. With the exception of 29PNS-80C, these environmentally qualified switches (Model No. EA740-80100) were installed in 1982 (29PNS-80C was installed in 1990). A review of the Plant Records Management data base shows that since 1982 all of the position switches have been replaced at least once. Some of the replacements were due to EQ requirements while other replacements were tied to poor performance. Typical problems involving the 10% closure limit switches include: failure of the limit switch to pick up, failure to reset, and slow resets. A review of work requests and operating occurrence reports associated with the MSIV limit switches revealed these failures are primarily being detected on-line. This evaluation also showed that most of the position switch deficiencies were related to reset of the switches rather than instrument drift, and therefore would not preclude extending the calibration interval.

A review of past test results for MST-29.2 was performed to determine whether individual drift values for a given calibration interval were within acceptable limits (MST-29.2 specifies a calibration tolerance for the valve position switches of ± 1% of valve closure). Calibration data from 1987-1992 was evaluated to establish actual past instrument drift. This review confirmed that past drift values for the limit switches were within the specified calibration tolerance except on rare occasions.

Field calibration data was also used to predict future instrument drift. The maximum expected drift for 30 months (MED30) was conservatively predicted to be 2.1% of valve closure. The existing trip setpoint for the MSIV position switches is 8% (± 1.0%) of valve closure. This trip setting provides a 2% margin between the trip point and the Technical Specification limit (Allowable Value) of $\leq$ 10% valve closure to account for instrument loop inaccuracies. Applying MED30 to the instrument loop shows that the Technical Specification trip setting could be exceeded with a 30 month calibration interval.

Based on the nature of the limit switch failures, and discussions with field personnel, changing the trip setpoint to 7% valve closure in an effort to bound all channel uncertainties would not be practical. Thus,

it is recommended that the Technical Specification trip setting be changed from $\leq$ 10% valve closure to $\leq$ 12.5% valve closure. This setting is consistent with NEDC-31336, "General Electric Instrument Setpoint Methodology," (Reference 45) which recommends a Technical Specification limit of 87.4% of full open position (12.6% closed). Revising the Tech Spec limit would then allow the limit switch trip setpoint to be changed from 8% to 10% of valve closure (see Table 2). This setting is more desirable for two reasons:

1) The revised settings would provide 2.5% margin between the Tech Spec limit (12.5%) and the field trip setting (10%) to account for instrument drift and loop inaccuracies. As a result, the calibration interval for the MSIV limit switches could be safely extended with the longer operating cycle since this margin accommodates uncertainties associated with a 30 month calibration interval.

2) The majority of the position switch problems were related to a failure of the switches to reset when the MSIV was re-opened following the functional test. Failure to reset is a recurrent problem caused by insufficient margin between the trip setpoint and the "reset point" of the limit switches. Revising the Tech Spec limit and the field trip setting will increase this margin thereby reducing the number of test failures and limit switch replacements.

In summary, the MSIV 10% closure limit switch calibration surveillance can be safely extended with the longer operating cycle. Sufficient on-line testing is provided to detect instrument failures. A review of maintenance work history and operating occurrence reports (Reference 2) for the 10% closure limit switches reveal that failures of the switches are primarily being detected through monthly and daily surveillances. In addition, conservative projections of instrument drift have been applied to the instrument loop. Revisions to the field trip setting and Technical Specification trip setting ensure that the calibration interval can be safely extended to a maximum of 30 months in addition to improving limit switch reliability.

11) Main Steam Line and Steam Jet Air Ejector Radiation Monitor Calibration (PSP-14, ISP-64-3)

The Main Steam Line Radiation Monitor (MSLRM) system consists of four redundant radiation detectors located external to the main steam lines outside of primary containment. The monitors are designed to detect a gross release of fission products indicative of fuel failure. The MSLRM currently provides readout, alarm and trip functions upon

detection of excessive radiation levels. A trip initiates a reactor scram, isolates the mechanical vacuum pumps, and initiates a Group 1 primary containment isolation signal for the Main Steam Isolation Valves (MSIV), main steam line drain valves, and recirculation loop sample valves.

Presently, the FitzPatrick Technical Specifications require a quarterly channel alignment of the Reactor Protection System circuitry associated with the main steam line high radiation trip using a standard current source. Calibration of the sensor using a radiation source is required once per operating cycle (Reference 40).

On July 15, 1993, NYPA submitted an application for an Amendment to the Technical Specifications (Reference 68) proposing elimination of the reactor scram and main steam isolation valve closure requirements associated with the main steam line radiation monitors (It is planned to retain the MSLRM isolation function for the main steam drain valves, the recirculation loop sample valves and the mechanical vacuum pumps isolation valves). Technical Specification Amendment 207 approved this change.

Elimination of the MSLRM scram function has been endorsed by the BWR Owners Group, and removes a potential source of spurious reactor transients. In addition, elimination of this scram signal does not result in an increased safety risk since automatic reactor shutdown on the MSLRM trip is not credited in the analysis of any design basis event.

PSP-14 calibrates the radiation detectors (17RE-230A through D) and evaluates the response of the entire instrument loop to a known radioactive source. Extension of the calibration interval for the radiation detectors is evaluated in Report No. JAF-RPT-PRM-01149, "Radiation Monitoring System Surveillance Test Extensions," (Reference 69). Once/3 months, ISP-64-3 (Reference 18) calibrates recorder 17RR-252 and radiation monitors 17RM-251A through D. This procedure also verifies that the RPS and Primary Containment Isolation System initiate a single channel trip when a test signal is applied. Following completion of the MSLRM elimination circuitry modifications, ISP-64-3 will require significant revision.

Based on the preceding discussion, further evaluation of the main steam line radiation monitor calibration as it applies to the Reactor Protection System is not required.

V. Surveillance Test Extensions

Response Time Testing

The measurement of response time provides assurance that the Reactor Protection System trip functions are completed within the time limits assumed in the transient and accident analyses. At FitzPatrick, response time credit is taken for the following Reactor Protection System parameters:

- APRM Flow Reference Simulated Thermal Power
- APRM Fixed Neutron Flux
- High Reactor Pressure
- Reactor Low Water Level
- Main Steam Isolation Valve Closure
- Turbine Stop Valve Closure
- Turbine Control Valve Fast Closure

In terms of the transient analysis an individual parameter response time is the time from when the parameter just exceeds its trip setpoint until the scram solenoids are de-energized (i.e 05A-K14 scram contacts open). The total time delay can be measured in any series of steps which, when added comprise the total response time as defined above.

Individual sensor response time is defined as the maximum allowable time from when the parameter being measured just exceeds the trip setpoint to opening of the channel sensor contact during a transient. The individual sensor response time is measured by simulating a step change of the parameter being measured. When sensor response time is measured independently, it is necessary to also measure the remaining portion of the response time in the logic train up to the time at which the scram pilot valves de-energize. The total channel response time includes: 1) all component delays in the response chain to the ATTS output relay (referred to as individual sensor response time), plus 2) the 50 ms design allowance for the RPS logic system response time (output of the MTU through the scram solenoids). A response time for the RPS logic relays in excess of 50ms is acceptable provided the overall response time does not exceed the Tech Spec limits as stated on Table 3.1-2.

The FitzPatrick Technical Specifications require that the response time for each reactor protection system trip function listed in Table 3.1-2 be demonstrated within the specified limits during each 18 month test interval. Each test includes at least one channel in each trip system. All channels in both trip systems must be tested within two test intervals.

The following RPS response time tests were evaluated to determine whether the surveillance test interval could be extended to support a 24 month operating cycle:

24 MONTH OPERATING CYCLE
REACTOR PROTECTION SYSTEM
SURVEILLANCE TEST EXTENSIONS

1) RPS reactor level instrument response time test (ISP-102)
2) RPS reactor pressure instrument response time test (ISP-104)
3) RPS drywell pressure instrument time response test (ISP-107)
4) RPS APRM response time test (ISP-108)
5) RPS main turbine control valve fast closure response time test (ISP-109)
6) RPS turbine stop valve closure response time test (ISP-110)
7) RPS MSIV less than 90% open response time test (ISP-111)

The primary significance of RPS response time testing is to periodically validate the scram insertion times assumed in the accident analyses. Testing of individual sensor response time was recently added to the Technical Specifications tc supplement the existing requirement for logic system response time verification. Inclusion of ATTS component response times in the total time delay ensures that transient and accident analyses assumptions are met.

Response time testing of RPS functions can be extended to support a 24 month operating cycle. Refueling interval calibrations and on-line surveillance testing (channel checks, channel functional tests and logic system functional tests) provide adequate assurance that these instrument channels are functioning as required. In addition, these surveillances would detect equipment failures which would significantly affect reactor protection system response times.

Each RPS parameter response time (sensor response time through de-energization of the scram solenoids) is actually a small fraction of the total allowable system response time for scram insertion. As a result, any postulated failures which result only in a slightly slower RPS response time are accommodated by the margins associated with the overall limits on scram insertion time.

Refueling is the practical time to perform response time testing. Complex response time testing requires stationing personnel at the location of instrument sensors where radiation exposure is significant. Consequently, response time testing of RPS instrument channels should be performed when there is access to containment.

It should be noted that response time testing has been used to detect failures of Rosemount transmitters susceptible to a slow loss of fill oil. Industry practice has shown however that drift analysis is the preferred method to detect the change in instrument performance for sensors which are prone to this type of failure (Reference 77). Furthermore, an evaluation of Rosemount transmitters (models 1151, 1152, 1153 and 1154) installed at the Fitzpatrick plant revealed that all Rosemount devices used in the Reactor Protection System are exempt from additional sensor monitoring requirements based on the criteria

established in NRC Bulletin No. 90-01, Supplement 1 (Reference 78)

Response time testing of Reactor Protection System trip functions can be safely extended with the longer operating cycle because: 1) refueling interval calibrations and on-line testing provide adequate assurance that equipment failures affecting RPS response time would be detected, 2) any postulated failures resulting in a slightly lower RPS response time could be accommodated by the margins associated with the overall limits on scram insertion time, and 3) performing these tests while the plant is on-line could result in unnecessary radiation exposure to plant personnel.

Refueling Interval Functional Tests

The following surveillance tests were also evaluated to determine whether the surveillance interval could be safely extended to support a 24 month operating cycle:

1) Mode Switch in Shutdown Functional Test (ST-29B)

The purpose of this test is to demonstrate:

- the ability of the reactor mode switch to cause a reactor scram when the switch is placed in shutdown
- operability of the time delay reset relays 5A-K22A and 5A-K22B
- capability of the scram discharge volume vent and drain valves to close in less than 30 seconds upon receipt of a scram signal, and open when the scram signal is reset

Verification that the SDIV vent and drain valve close in less than 30 seconds is addressed in the system report for Control Rod Drive Surveillance Test Extensions (Reference 79). Justification for extension of the test interval for the reactor mode switch and reset relays are discussed below.

Placing the mode switch in shutdown actuates the scram relays (5A-K15) on the manual trip channels A3 and B3. The mode switch and its associated trip logic are functionally tested during each refueling outage by placing the mode switch in shutdown and verifying proper RPS response. Operability of the scram reset relays (5A-K22A&B) is verified following the reactor scram by depressing the Rx Scram Reset switch and recording the time it takes for the actuators to reset.

The manual scram trip logic and reset relays are functionally tested at quarterly intervals by surveillance test ST-29A (Reference 12). In this test each manual scram pushbutton is depressed and half scram indications are observed. The scram is then reset and indications are observed to be clear. This test provides assurance that the trip logic actuated by placing the mode switch in shutdown is functioning properly as well as verifying operability of the scram reset relays. The reactor mode switch is also used to shutdown the reactor for scheduled maintenance outages. As a result, failures of the 5A-K15 scram relays and the reset relays would be detected on-line. Since operability of the scram circuitry and reset relays is verified on-line, the primary objective of the refueling interval test is to check mode switch functionality and the reset relays time delay.

The mode switch in service at FitzPatrick is a General Electric Type SB-9 switch designed for highly repetitive service. This switch was installed in 1987 as a replacement for the SB-1 switch which was previously in use. The switch was replaced based on recommendations contained in GE Service Information Letter (SIL) 397 (Reference 80), after a SB-1 switch malfunctioned at an operating plant. The SIL recommended that the SB-1 switch be replaced with the longer life SB-9 switch which is designed for heavy duty operation.

In September of 1989, GE issued another SIL related to reactor mode switches Types SB-1 and SB-9 (Reference 81). This letter reported two failures of a particular lock handle used on reactor mode switches. The reported failures were incomplete transfer from one lock handle position to another. SIL 498 provided recommended action for all BWR owners using type SB-1 and SB-9 control switches with Corbin lock handles. In August 1990, the reactor mode switch at FitzPatrick was found to be loose with lateral movement. The mechanism was reworked per SIL 498 prior to reactor start-up (Reference 82) and retested satisfactorily. Since that time, a review of the work request data bases and operating occurrence reports show no additional failures of this switch.

Surveillance test ST-29B also verifies that the time delay for the reset relays is greater than or equal ($\geq$) to 10 seconds. A review of test results from 1987 - 1992 show that the reset time always exceeded 10 seconds. As a result, past performance of the reset relay time delay function has been satisfactory.

The mode switch in shutdown functional test can be safely extended with the longer operating cycle because: 1) on-line

testing verifies operability of the manual scram circuitry and the reset relays, 2) since rework of the mode switch (per GE Service Information Letters) there have been no problems with the reactor mode switch, and 3) a review of past test results shows acceptable values for the reset relays time delay.

2) RPS Channel Test Switch Functional Test (ST-29C)

Functional testing of the RPS channel test switches is required by the Technical Specifications (Table 4.1-1 #3) every refueling outage or after channel maintenance. This test demonstrates operability of RPS trip channels A and B by verifying the ability of each logic train (A1, A2, B1, B2) to de-energize the actuators for each trip system. In this test, the keylock test switches (5A-S2A through D) are placed in the TRIP position one at a time. Actuation of each trip systems scram relays (5A-K14 relays) is verified by observing scram group lights, annunciators and EPIC.

The RPS channel test switch functional test can be safely extended with the longer operating cycle. On-line testing provides adequate assurance that the scram relays are functioning properly. These relays are exercised weekly, and monthly by channel functional tests for the various automatic scram signals. As a result, the primary purpose of the refueling interval surveillance (ST-29C) is to verify the functionality of the test switch.

The RPS channel test switches are General Electric Type CR2940 devices. A review of the work request data base and the plant records management system indicates that there have been no problems with these switches. In addition, a review of past test results (from 1987 - 1992) and operating occurrence reports shows no failures of either surveillance test ST-29C or of the switches themselves.

In summary, the channel test switch functional test for the RPS can be safely extended with the longer operating cycle. The primary objective of this test is to demonstrate operability of each trip systems scram relays. These relays are exercised on-line by channel functional tests for the automatic scram signals. Furthermore, a review of past test results, occurrence reports and work request data bases revealed that the switches have been reliable.

3) Backup Scram Valves Functional Test (ST-29E)

The backup scram valves provide a second method of venting the air pressure from the scram valves in the event either scram pilot valve solenoid for any control rod fails to de-energize when a scram is required. The DC solenoid for each backup scram valve is normally de-energized. The backup scram valves are energized when both trip systems A and B are tripped. This action initiates insertion of every control rod regardless of the action of the scram pilot valves.

The purpose of this test is to demonstrate the operability of backup scram valves 03SOV-140A and 03SOV-104B by individually energizing each valve and observing a reduction in scram air header pressure. Functional testing of the backup scram valves is not required by the Technical Specifications. This test was added to the surveillance program as a result of NRC Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events" (Reference 83). Generic Letter 83-28 required licensees to implement on-line functional testing of the reactor trip system, including independent testing of the diverse trip features when possible. These requirements include on-line surveillance testing of the scram pilot valves and backup scram valves (including all initiating circuitry) for General Electric Plants.

The backup scram design at FitzPatrick does not permit performance of on-line functional testing of the backup scram valves. This is documented in NYPAs response to Generic Letter 83-28 (Reference 84). As an alternative to on-line testing of backup scram valves, NYPA implemented functional testing of these valves once each refueling cycle while the plant is shutdown. Because of the system design, refueling is the practical time to perform functional testing of the backup scram valves, while functional testing of the initiating circuitry is performed at weekly and monthly intervals. This surveillance frequency is consistent with industry practices.

A review of operating occurrence reports and work request data bases was completed to assess the reliability of the backup scram valves. This evaluation revealed no problems with either of the backup scram valves (03SOV-140A&B) for the past five years. During the test performed on May 20, 1990, however, a full scram was received while performing the first section of ST-29E. This problem occurred when the half scram did not reset as per procedure and a full scram was inserted when the jumper was removed to restore the system. There was no malfunction of the

backup scram valves during the test, and an investigation of the failure of the half scram to reset was completed. Based on this review, it is concluded that the backup scram valves have had reliable past performance and failures of these valves are not expected with the longer operating cycle.

Test requirements for the backup scram valves are not specified in the plant Technical Specifications. In addition no credit is taken for these valves in the FitzPatrick safety analyses since all of the transient and accident analyses assume normal scram function. As a result, the backup scram valves only enhance the scram reliability of the reactor protection system.

The backup scram valves functional test can be safely extended with the longer operating cycle because: 1) the backup scram design at FitzPatrick does not permit performance of on-line testing of the valves, 2) refueling is the practical time to perform this surveillance and is consistent with standard industry practice, 3) a review of operating occurrence reports and work request data bases did not reveal any problems with these valves, and 4) functional testing of the backup scram valves is not required by the Technical Specifications and no credit is taken for them in the safety analysis.

VI. Summary and Conclusions

To meet the requirements of NRC Generic Letter 91-04, a plant specific instrument drift analysis was completed for RPS circuits affected by an increased calibration interval (i.e. refueling interval tests/calibrations). The result of the instrument drift evaluation is a predicted "as-found" value for the components which comprise the instrument loop. Generic Letter 91-04 also requires that these projections of instrument drift be consistent with the values of drift errors used in loop accuracy/setpoint calculations. Postulated values of future instrument drift were incorporated into loop accuracy calculations for the appropriate RPS circuit. Based on the conclusions of the instrument drift evaluation and the loop accuracy calculations one of the following applies for each RPS function evaluated:

a) Sufficient margin exists between the field trip setpoint and the analytical limit to accommodate postulated increases in drift and uncertainties associated with the longer calibration interval; or

b) Instrument drift for a 30 month calibration interval can be safely accommodated provided a change is made to the existing field trip setpoint and/or the Technical Specification trip level setting; or

c) Additional calibration data is required to support extension of the

calibration interval at this time (note: applies to the APRM flow bias circuit).

Evaluation of calibration interval extensions also considers on-line channel checks, channel functional tests and on-line calibrations of master/slave trip units. These surveillances would detect significant drift or circuit failures.

An evaluation of RPS surveillance test results, calibration data and loop accuracy calculations shows that the following instrument calibrations can be safely extended to a maximum of 30 months:

1. SDIV High Water Level Instrument Functional Test/Calibration (ISP-66-1)
2. SDIV Water Level Transmitter Calibration (ISP-66-4)
3. Turbine Control Valve Oil Pressure (ISP-69-1)
4. RPS and PCIS Analog Trip System Master and Slave Trip Unit Alignment (ISP-101A through D)
5. RPS-PCIS Pressure Transmitter Calibration (ISP-200A through D)

In addition, calibration intervals for the following procedures can be safely extended provided changes are made to the field trip setpoints and/or Tech Spec trip settings:

1. Turbine Stop Valve Closure Instrument Calibration (ISP-37)
2. RPS Electrical Protection Assembly Functional Test/Calibration (ISP-94)
3. RPS-PCIS Reactor Level Transmitter Calibration and Channel Functional Test (ISP-201A through D)
4. Main Steam Line Isolation Valve 10% Closure Limit Switch Calibration Surveillance (MST-29.2)

Proposed changes to the affected RPS setpoints are included in Table 2. These changes provide assurance that system safety limits would not be exceeded for the duration of the longer operating cycle. Table 3 provides recommended changes to calibration procedures. These modifications will better reflect true instrument performance (based on vendor specifications) for a 30 month calibration interval. In addition, the procedural changes will provide a means of detecting instruments which are performing poorly.

An extension of the following "once per refueling" surveillances to accommodate a 24 month operating cycle is also justified:

- Response time tests for the Reactor Protection system (ISP-102, 104, 107, 108, 109, 110, and 111) can be safely extended with the longer operating cycle because: 1) refueling interval calibrations and on-line testing provide adequate assurance that equipment failures affecting

RPS response times would be detected, 2) postulated failures resulting in slightly lower response times can be accommodated by the margins associated with the overall limits on scram insertion time, and 3) performing these tests on-line could result in unnecessary radiation exposure to plant personnel.

- The Mode switch in shutdown functional test (ST-29B) can be safely extended because: 1) on-line testing verifies operability of the manual scram circuitry and the reset relays, 2) there have been no problems with the reactor mode switch since rework of the mode switch per GE service information letters, and 3) a review of past test results shows acceptable values for the reset relays time delay function.

- The RPS channel test switch functional test (ST-29C) can be extended with the longer operating cycle because the scram relays are exercised on-line by the various channel functional tests for the automatic scram signals. In addition, a review of past test results, operating occurrence reports and work request data bases revealed that the channel test switches have been reliable.

- The backup scram valves functional test (ST-29E) can be safely extended with the longer operating cycle because: 1) the backup scram design at FitzPatrick does not permit performance of on-line testing of the valves, 2) refueling is the practical time to perform this surveillance and is consistent with standard industry practice, 3) a review of operating occurrence reports and work request data bases did not reveal any problems with these valves, and 4) functional testing of the backup scram valves is not required by the Technical Specifications and no credit is taken for them in the safety analysis.

Our evaluations also conclude that insufficient data is available to support extension of the calibration interval for the APRM flow biasing instrument circuit. It is suggested that instrument performance of the Rosemount flow transmitters be monitored. Extension of the calibration interval for the new transmitters will be justified provided a pattern of repeatable, acceptable drift values is established.

Reactor Protection System surveillance test/calibration extensions require an amendment to the FitzPatrick Technical Specifications. Marked-up Technical Specification pages and a supporting Safety Evaluation are included in Attachments 1 and 2.

VII. References

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2. Operational Occurrence Report Logs from 1987 to 1992.

3. James A. FitzPatrick Nuclear Power Plant, Updated Final Safety Analysis Report, Sections 7.2 and 7.5.

4. James A. FitzPatrick Nuclear Power Plant, Technical Specifications Sections 3.1/4.1, 2.1 and 3.9.G/4.9.G.

5. James A. FitzPatrick Nuclear Power Plant Operating Procedure OP-18, "Reactor Protection System," Revision 14, dated May 24, 1991.

6. Operations Surveillance Test Procedure No. 40D, "Daily Surveillance and Instrument Check", Revision 67, dated May 13, 1994.

7. Operations Surveillance Test Procedure No. 1E, "Main Turbine Stop Valve Limit Switch Instrument Functional Test", Revision 8, dated March 19, 1993.

8. Operations Surveillance Test Procedure No. 1G, "Main Steam Line High Radiation Functional Test", Revision 14, dated June 13, 1991.

9. Operations Surveillance Test Procedure No. 1I, "Main Steam Isolation Valves Limit Switch Instrument Functional Test", Revision 16, dated May 19, 1993.

10. Operations Surveillance Test Procedure No. 1L, "Main Turbine Control Valve Instrument Channel and Valve Operability Check", Revision 17, dated March 3, 1993.

11. Operations Surveillance Test Procedure No. 5Q, "APRM Flow Bias Functional Test (Run Mode)", Revision 10, dated March 22, 1993.

12. Operations Surveillance Test Procedure No. 29A, "Manual Scram Functional Test", Revision 8, dated September 2, 1992.

13. Operations Surveillance Test Procedure No. 29B, "Mode Switch in Shutdown Functional Test", Revision 13, dated September 18, 1992.

14. Operations Surveillance Test Procedure No. 29C, "RPS Channel Test Switch Functional Test", Revision 6, dated September 4, 1992.

15. Operations Surveillance Test Procedure No. ST-29E, "Backup Scram

Valves Functional Test," Revision 4, dated September 4, 1992.

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17. I&C Surveillance Procedure No. 63-1, "APRM Flow Bias Signal Instrument Calibration", Revision 12, dated November 6, 1992.

18. I&C Surveillance Procedure No. 64-3, "Main Steam Radiation Monitor Instrument Calibration", Revision 8, dated September 18, 1992.

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21. I&C Surveillance Procedure No. 69-1, "Turbine Control Valve Oil Pressure", Revision 9, dated September , 1989.

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24. I&C Surveillance Procedure No. 100B, "Reactor Protection System and Primary Containment Isolation System Instrument Functional Test/Calibration (ATTS)", Revision 14, dated August 21, 1992.

25. I&C Surveillance Procedure No. 100C, "Reactor Protection System and Primary Containment Isolation System Instrument Functional Test/Calibration (ATTS)", Revision 16, dated August 20, 1992.

26. I&C Surveillance Procedure No. 100D, "Reactor Protection System and Primary Containment Isolation System Instrument Functional Test/Calibration (ATTS)", Revision 21, August 21, 1992.

27. I&C Surveillance Procedure No. 101A, "RPS and PCIS Analog Trip System Master Trip Unit Alignment", Revision 4, dated October 24, 1991.

28. I&C Surveillance Procedure No. 101B, "RPS and PCIS Analog Trip

System Master Trip Unit Alignment", Revision 3, dated October, 1988.

29. I&C Surveillance Procedure No. 101C, "RPS and PCIS Analog Trip System Master and Slave Trip Unit Alignment", Revision 2, dated October, 1988.

30. I&C Surveillance Procedure No. 101D, "RPS and PCIS Analog Trip System Master and Slave Trip Unit Alignment", Revision 3, dated October, 1988.

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37. I&C Surveillance Procedure No. 201C, "RPS-PCIS Pressure Level Transmitter Calibration and Channel Functional Test (ATTS)", Revision 5, dated January 6, 1993.

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Radiation Monitor Calibration," Revision 6, dated August 14, 1992.

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51. JAF-CALC-CRD-00234, "Setpoint Calculation for 03DPT-231C,D - Discharge Volume Scram," Revision 1, dated June 10, 1992.

52. JAF-CALC-CRD-00307, "Setpoint Calculation for 03DPT-231G,H - Discharge Volume Scram," Revision 1, dated June 10, 1992.

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54. JAF-CALC-MTG-00667, "Setpoint Calculation for 94PS-20A,B,C&D - Turbine Control Fast Closure," Revision 0, dated May 20, 1993.

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ATTACHMENT 1

TECHNICAL SPECIFICATION CHANGES

2.1 (cont'd)

2. Reactor Water Low Level Scram Trip Setting

Reactor low water level scram setting shall be ≥177 in. above the top of the active fuel (TAF) at normal operating conditions.

3. Turbine Stop Valve Closure Scram Trip Setting

Turbine stop valve scram shall be ≤10 percent valve closure from full open when above 217 psig turbine first stage pressure.

4. Turbine Control Valve Fast Closure Scram Trip Setting

Turbine control valve fast closure scram control oil pressure shall be set at 500 < P < 850 psig.

5. Main Steam Line Isolation Valve Closure Scram Trip Setting

Main steam line isolation valve closure scram shall be ≤ ~~10~~ 12.5 percent valve closure from full open.

6. Main Steam Line Isolation Valve Closure on Low Pressure

When in the run mode main steam line low pressure initiation of main steam line isolation valve closure shall be ≥825 psig.

3.1 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the instrumentation and associated devices which initiate the reactor scram.

Objective:

To assure the operability of the Reactor Protection System.

Specification:

A. The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch, shall be as shown in Table 3.1-1. The reactor protection system instrumentation response time shall be within the limits in Table 3.1-2.

B. Minimum Critical Power Ratio (MCPR)

During reactor power operation, the MCPR operating limit shall not be less than that shown in the Core Operating Limits Report.

1. During Reactor power operation with core flow less than 100% of rated, the MCPR operating limit shall be multiplied by the appropriate K_f as specified in the Core Operating Limits Report.

4.1 SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective:

To specify the type of frequency of surveillance to be applied to the protection instrumentation.

Specification:

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1-1 and 4.1-2 respectively.

The response time for each reactor protection system trip function listed in Table 3.1-2 shall be demonstrated to be within the limits in the table ~~during each 18 month test interval~~. Each test shall include at least one channel in each trip system. All channels in both trip systems shall be tested within two test intervals.

at least once every 24 months

(ISP-102, 104, 107, 108, 109, 110, 111)

B. Maximum Fraction of Limiting Power Density (MFLPD)

The MFLPD shall be determined daily during reactor power operation at ≥25% rated thermal power and the APRM high flux scram and Rod Block trip settings adjusted if necessary as specified in the Core Operating Limits Report.

TABLE 3.1-1 (cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENT

Minimum No. of Operable Instrument Channels per Trip System (1)	Trip Function	Trip Level Setting [1]	Modes in Which Function Must be Operable			Total Number of Instrument Channels Provided by Design for Both Trip Systems	Action (1)
			Refuel (6)	Startup	Run		
2	APRM Downscale	≥2.5 indicated on scale (9)			X	6 Instrument Channels	A or B
2	High Reactor Pressure	≤1045 psig	X(8)	X	X	4 Instrument Channels	A
2	High Drywell Pressure	≤2.7 psig	X(7)	X(7)	X	4 Instrument Channels	A
2	Reactor Low Water Level	≥177 in. above TAF	X	X	X	4 Instrument Channels	A
3	High Water Level in Scram Discharge Volume	≤34.5 gallons per Instrument Volume	X(2)	X	X	8 Instrument Channels	A
2	Main Steam Line High Radiation	≤3x normal full power background (16)	X	X	X	4 Instrument Channels	A
4	Main Steam Line Isolation Valve Closure	≤ ~~10%~~ 12.5% valve closure			X(5)	8 Instrument Channels	A

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Table 4.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT FUNCTIONAL TEST

MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY INSTRUMENT AND CONTROL CIRCUITS

Instrument Channel	Group	Functional Test	Minimum Frequency (3)
Mode Switch in Shutdown	A	Place Mode Switch in Shutdown	~~Each refueling outage.~~ At least once/24 month (SI-298)
Manual Scram	A	Trip Channel and Alarm	Every 3 months.
RPS Channel Test Switch	A	Trip Channel and Alarm	At least once/24 month ~~Every refueling outage~~ or (SI-298 after channel maintenance.
IRM High Flux	C	Trip Channel and Alarm (4)	Once per week during refueling or startup and before each startup.
IRM Inoperative	C	Trip Channel and Alarm (4)	Once per week during refueling or startup and before each startup.
APRM			
High Flux	B	Trip Output Relays (4)	Once/week.
Inoperative	B	Trip Output Relays (4)	Once/week.
Downscale	B	Trip Output Relays (4)	Once/week.
Flow Bias	B	Calibrate Flow Bias Signal (4)	Once/month (1).
High Flux in Startup or Refuel	C	Trip Output Relays (4)	Once per week during refueling or startup and before each startup.
High Reactor Pressure	B	Trip Channel and Alarm (4)	Once/month. (1)(8)
High Drywell Pressure	B	Trip Channel and Alarm (4)	Once/month. (1)(8)
Reactor Low Level	B	Trip Channel and Alarm (4)	Once/month. (1)(8)
High Water Level in Scram Discharge Instrument Volume	A	Trip Channel	Once/month. (7)
High Water Level in Scram Discharge Instrument Volume	B	Trip Channel and Alarm (4)	Once/month. (1)(8)

TABLE 4.1-2

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency (2)
IRM High Flux	C	Comparison to APRM on Controlled Shutdowns	Maximum frequency once/week
APRM High Flux Output Signal	B	Heat Balance	Daily
Flow Bias Signal	B	Internal Power and Flow Test with Standard Pressure Source	At least once every 18 months (ISP 63 1) ~~Every refueling outage~~
LPRM Signal	B	TIP System Traverse	Every 1000 effective full power hours
High Reactor Pressure	B	Standard Pressure Source	Note (6) (ISP - 200A, B, C, D)
High Drywell Pressure	B	Standard Pressure Source	Note (6) (ISP 200A, B, C, D)
Reactor Low Water Level	B	Standard Pressure Source	Note (6) (ISP 201A, B, C, D)
High Water Level in Scram Discharge Instrument Volume	A	Water Column, Note (5)	At least once/24 months (ISP-66 1) ~~Once/operating cycle, Note (5)~~
High Water Level in Scram Discharge Instrument Volume	B	Standard Pressure Source	At least once every 24 months (ISP 66 4) ~~Every 3 months~~
Main Steam Line Isolation Valve Closure	A	Note (4)	Note (4) (MST-29.2)
Main Steam Line High Radiation	B	Standard Current Source (3)	Every 3 months (ISP-64-3, PSP-14)
Turbine First Stage Pressure Permissive	B	Standard Pressure Source	Note (6) (ISP-200A, B, C, D)

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TABLE 4.1-2 (Cont'd)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group (1)	Calibration	Minimum Frequency (2)	
Turbine Control Valve Fast Closure Oil Pressure Trip	A	Standard Pressure Source	At least once/24 months ~~Once/operating cycle~~	(ISP 49-1)
Turbine Stop Valve Closure	A	Note (4)	Note (4)	(ISP 37)

NOTES FOR TABLE 4.1-2

1. A description of three groups is included in the Bases of this Specification.
2. Calibration test is not required on the part of the system that is not required to be operable, or is tripped, but is required prior to return to service.
3. The current source provides an instrument channel alignment. Calibration using a radiation source shall be performed each refueling outage.
4. Actuation of these switches by normal means will be performed ~~during the refueling outages~~. at least once/24 months.
5. Calibration shall be performed utilizing a water column or similar device to provide assurance that damage to a float or other portions of the float assembly will be detected.
6. Sensor calibration at least once/24 months ~~once per operating cycle~~. Master/slave trip unit calibration once per 6 months.

3.9 (cont'd)

3. From and after the time both power supplies are made or found inoperable the reactor shall be brought to cold condition within 24 hours.

G. REACTOR PROTECTION SYSTEM ELECTRICAL PROTECTION ASSEMBLIES

Two RPS electrical protection assemblies for each inservice RPS MG set and inservice alternate source shall be operable except as specified below:

1. With one RPS electrical protection assembly for an inservice RPS MG set or an inservice alternate power supply inoperable, restore the inoperable channel to operable status within 72 hours or remove the associated RPS MG set or alternate power supply from service.

2. With two RPS electrical protection assemblies for an inservice RPS MG set or an inservice alternate power supply inoperable, restore at least one to operable status within 30 minutes or remove the associated RPS MG set or alternate power supply from service.

4.9 (cont'd)

G. REACTOR PROTECTION SYSTEM ELECTRICAL PROTECTION ASSEMBLIES

The RPS electrical protection assemblies instrumentation shall be determined operable by:

1. Performing a channel functional test each time the plant is in cold shutdown for a period of more than 24 hours, unless performed in the previous 6 months.

2. At least once ~~per operating cycle~~ every 24 months, demonstrating the operability of over-voltage, under-voltage and under-frequency protective instrumentation by performance of a channel calibration including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following set points:

	RPS MG SET SOURCE
OVER-VOLTAGE	≤132V ≤~~4~~ 3.2 second Time Delay
UNDER-VOLTAGE	~~≥108V~~ ≥111.4V ≤~~4~~ 3.1 second Time Delay
UNDER-FREQUENCY	≥57HZ ≤~~4~~ 2.7 second Time Delay

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(continued on page 222d)

3.9.G (cont'd)

3. With the reactor in the RUN mode, at least one (1) RPS division shall be powered from the MG set except as specified below:

   - With both RPS divisions powered from the alternate sources, at least one division power source shall be restored to a MG set with operable electrical protection assemblies within seven (7) days or the reactor shall be brought to the cold condition within the subsequent 24 hours.

4.9.G (cont'd)

	ALTERNATE SOURCE
OVER-VOLTAGE	≤132V ≤4s Time Delay
UNDER-VOLTAGE	≥108V ≤4s Time Delay
UNDER-FREQUENCY	≥57HZ ≤4s Time Delay

111.4V

Revision 1

ATTACHMENT 2

SAFETY EVALUATION

24 MONTH OPERATING CYCLE
REACTOR PROTECTION SYSTEM
SURVEILLANCE TEST EXTENSIONS

I. Safety Evaluation

The proposed changes to RPS calibration and surveillance test intervals have been reviewed in accordance with the requirements of 10CFR 50.59 and 10CFR 50.92. These changes, which extend calibration and test intervals do not involve unreviewed safety questions, nor do they constitute a Significant Hazards Consideration.

1. *The probability of occurrence and the consequences of an accident or malfunction of safety related equipment previously evaluated in the safety analysis report will not be increased.*

   Changes are proposed to increase the surveillance test interval (STI) with the nominal 24 month fuel cycle for RPS functional tests and calibration of RPS instrument loops.

   These changes extend the STIs. They do not involve any hardware modifications. There is no increase in (1) the probability of an accident occurring, (2) the consequences of an accident, and (3) the consequences of equipment malfunction. However, increasing the STIs may affect the probability of equipment malfunction.

   Regarding the probability of equipment malfunction:

   - RPS calibration procedures can be safely extended with the longer operating cycle because a comparison of past and projected drift (for a 30 month cycle) with vendor specified uncertainties or the calibration tolerance support extension of the calibration interval. In addition, setpoint calculations confirm that sufficient margin exists between existing trip settings and the analytical limits to accommodate postulated drift or uncertainties associated with the longer refueling cycle. RPS instrumentation is also subjected to on-line surveillances including functional tests and channel checks. These surveillances provide assurance that RPS components are operating as required. The calibration interval for APRM flow bias instrumentation should not be extended at this time since there is insufficient data available to support this extension.

   - Response time tests for the Reactor Protection System can be safely extended with the longer operating cycle because: 1) refueling interval calibrations and on-line testing provide adequate assurance that equipment failures affecting RPS response times would be detected, 2) postulated failures resulting in slightly lower response times can be accommodated by the margins associated with the overall limits on scram insertion time, and 3) performing these tests on-line could result in

unnecessary radiation exposure to plant personnel.

- The Mode switch in shutdown functional test can be safely extended because: 1) on-line testing verifies operability of the manual scram circuitry and the reset relays, 2) there have been no problems with the reactor mode switch since rework of the mode switch per GE service information letters, and 3) a review of past test results shows acceptable values for the reset relays time delay function.

- The RPS channel test switch functional test can be extended with the longer operating cycle because the scram relays are exercised on-line by the various channel functional tests for the automatic scram signals. In addition, a review of past test results, operating occurrence reports and work request data bases revealed that the channel test switches have been reliable.

- The backup scram valves functional test can be safely extended with the longer refueling cycle because: 1) the backup scram design does not permit performance of on-line testing of the valves, 2) refueling is the practical time to perform this surveillance and is consistent with standard industry practice, 3) a review of operating occurrence reports and work request data bases did not reveal any problems with these valves, and 4) functional testing of the backup scram valves is not required by the Technical Specifications and no credit is taken for them in the safety analysis.

2. *The possibility of an accident or malfunction of a different type than evaluated previously in the safety analysis report is not created.*

   The proposed changes extend STIs. The proposed changes do not change the manner in which the Reactor Protection System functions. An evaluation of past equipment performance, instrument drift, existing trip settings and a study of on-line testing show the longer STIs will not degrade RPS equipment. Therefore, the proposed changes do not create any new failure modes or a new accident.

3. *The margin of safety as defined in the basis for any Technical Specification is not reduced.*

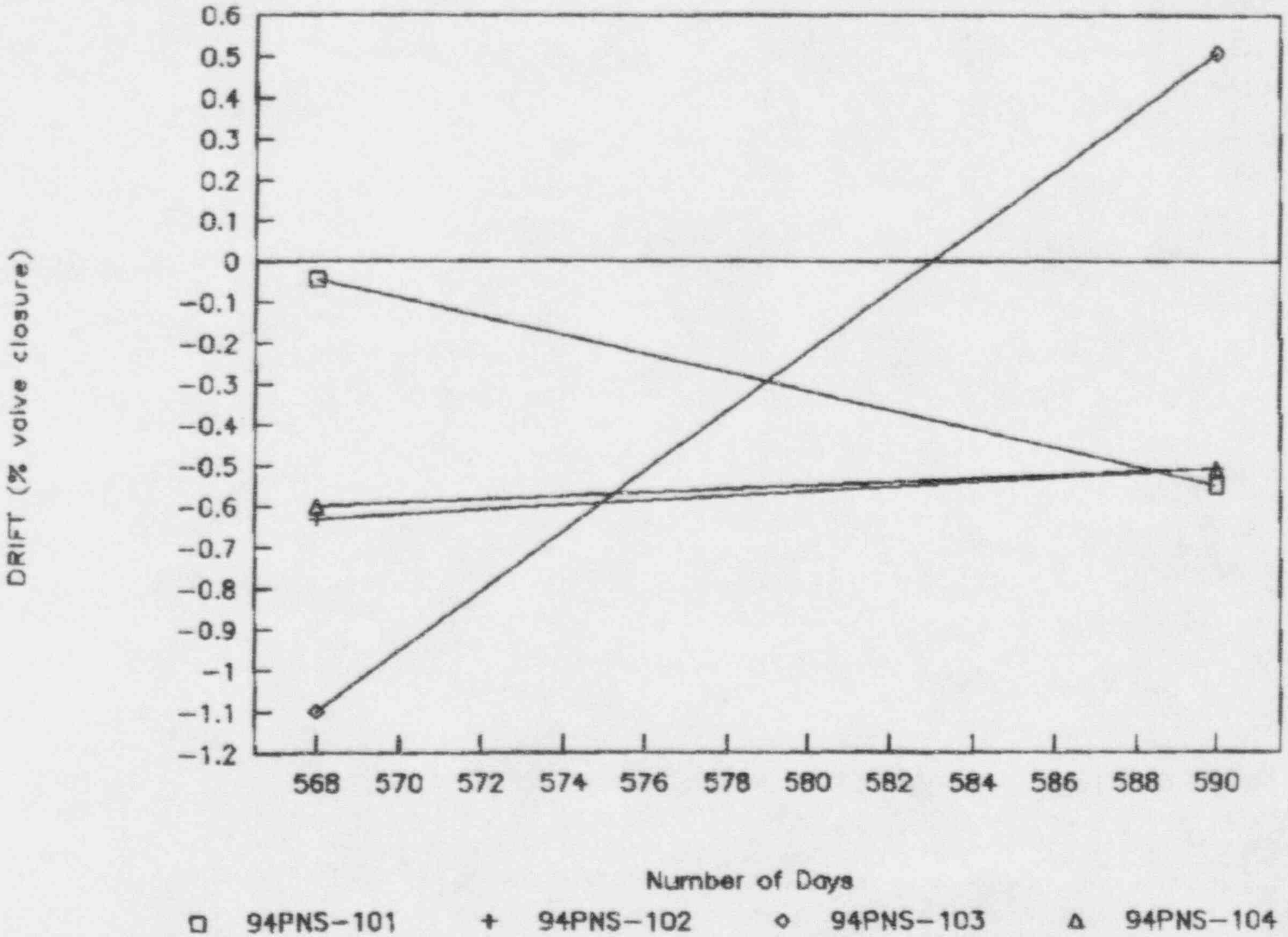
   The proposed changes do not reduce the margin of safety as defined in the basis for any Technical Specifications. The proposed changes extend STIs. Evaluation of the past performance of the equipment indicates that the effects of extending the STIs would not involve a significant reduction in the margin of safety.

FITZPATRICK - 24 MONTH OPERATING CYCLE

ATTACHMENT 3

DRIFT TREND FOR TSV POSITION SWITCHES

DRIFT TREND FOR TSV POSITION SWITCHES



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