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TECHNICAL EVALUATION REPORT ON THE NEUTRON DETECTOR DECALIBRATION AT THE FORT ST. VRAIN NUCLEAR GENERATING STATION

(Docket No. 50-267)

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Lawrence Livermore National Laboratory

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ABSTRACT

This report documents the technical evaluation on the decalibration of the neutron detectors at the Fort St. Vrain Nuclear Generating Station. The evaluation is to determine that the added circuitry for generating a floating trip setpoint as a function of indicated power meets NRC design criteria and has no adverse effects on the plant protection system.

The evaluation finds that the floating trip setpoint circuitry meets the design criteria specified in the plant's FSAR and will produce a reactor trip (as a function of indicated power) before the true power limit value in the Technical Specifications is exceeded.

FOREWORD

This report is supplied as part of the Selected Operating Reactor Issues Program II being conducted for the U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of Licensing, by Lawrence Livermore National Laboratory.

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TECHNICAL EVALUATION REPORT ON THE NEUTRON DETECTOR DECALIBRATION AT THE FORT ST. VRAIN NUCLEAR GENERATING STATION

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1. INTRODUCTION

The excore neutron detectors at the Fort St. Vrain (FSV) Nuclear Generating Station are located in the prestressed concrete reactor vessel adjacent to the core. The instrumentation inputs from these neutron flux detectors are used in reactor control and the plant's protection system (PPS).

The excore neutron detectors, 12 in number, are located in 6 wells at 60° intervals around the core cavity. The function of these twelve detectors (power range) are as follows:

(1) Six detectors are used in the PPS. The signals from these six are combined into three channels by two 180° opposing detectors. The range of these detectors is from 1.5% to 150% of full power. These three channels provide a trip signal in a 2-of-3 channel logic at 140% of full power.

(2) Six detectors are used in reactor control. They also have a range from 1.5% to 150% of full power. The signals from these detectors (flux controller) are used to regulate the position of the control rod pair and runback rods to control the power level in the core. The flux recorder, flux integrator (megawatt-hour meter), and power/flow module also receive input from the flux controller.

The neutron flux level as measured by the power range detectors is effected by the motion of the control rods. This motion can alter the radial core power distribution so that the flux levels measured are not directly proportional to the true core thermal power level thus indicating "decalibration" of the detectors.

Analyses have shown that motion of the rod banks near the center of the core causes the detectors to underpredict true power changes while rod banks near the outside cause overprediction of true power changes [Refs. 1 and 2]. The effects of this decalibration and resulting over/under predictions of true power could cause spurious trip signals or cause design limits to be exceeded before a protective trip occurs.

The rod withdrawal prohibits (RWPs) are also affected by detector decalibration. These RWPs are activated at 120% reactor power and at a power level that is not within the range as permitted by one of the three positions of the interlock sequence switch (ISS).

The effects of rod motion on decalibration of the detectors was analyzed by General Atomic Company (GAC). GAC recommended a circuit design change be implemented to provide for a "floating trip setpoint" to eliminate the decalibration problems due to rod motion. By letters dated January 11, 1979 [Ref. 1], November 29, 1979 [Ref. 2], May 16, 1983 [Ref. 3], and November 30, 1983 [Ref. 4], Public Service Company of Colorado, the licensee, submitted General Atomic Company's analysis and design reports and plant technical information on the proposed design change for a floating trip setpoint circuit at the Fort St. Vrain Nuclear Generating Station.

The purpose of this report is to evaluate the design modification of a floating trip setpoint circuit to determine that it meets NRC design criteria and will not adversely affect the operation of the PPS. The evaluation only included the instrumentation and control modification and did not cover reactor physics or core performance.

2. DESIGN REVIEW CRITERIA

The review criteria that were applied in determining the acceptability of the floating trip setpoint circuitry in the PPS are listed below. The existing reactor trip system design is based on this criteria.

(1) "AEC General Design Criteria for Nuclear Power Plant Construction Permits," 1967 edition [Ref. 5].

Criterion 7:	Suppression of Power Oscillations
Criterion 12:	Instrumentation and Control Systems
Criterion 15:	Engineered Safety Features Protection Systems
Criterion 19:	Protection Systems Reliability
Criterion 20:	Protection Systems Redundancy and Independence
Criterion 21:	Single Failure Definition
Criterion 22:	Separation of Protection and Control Instrumentation Systems
Criterion 23:	Protection Against Multiple Disability for Protection Systems
Criterion 24:	Emergency Power for Protection Systems
Criterion 25:	Demonstration of Functional Operability of Protection System
Criterion 26:	Protection Systems Fail-Safe Design
Criterion 39:	Emergency Power for Engineered Safety Features

(2) IEEE Standard: "Criteria for Protection Systems for Nuclear Power Generating Stations," IEEE 279-1968 [Ref. 6].

3. DESIGN DESCRIPTION

The floating trip setpoint circuit (FTSC) design is shown in Figures 1 through 5. These figures are taken from the General Atomic Company Report [Ref. 1].

The basic function of the FTSC is to produce a floating trip setpoint that will vary at a constant offset above indicated power which will always produce a trip signal before the reactor reaches the 140% true power value specified in the Technical Specifications. Figure 1 shows the basic electronic components which make up the floating trip setpoint circuitry (indicated by the dashed lines). The theory of operation can best be described using Figure 3. An indicated power signal derived from the power range detectors (0-150%) is fed to a differentiator, sample and hold (S/H) circuit, and to the two bistable trips (reactor and rod withdrawal prohibit).

The differentiator outputs a signal (volts) which is proportional to the rate of change of the indicated power. This output is fed to a bistable trip where it is compared to a pre-selected rate of power change. If the comparator goes low, the S/H circuit will continue to sample the indicated power input. Should the comparator go high, the S/H will hold its last input signal before the S/H goes high and feeds this value to a summer. At the summer, the input value from the S/H is added to the pre-selected trip offset value. If the sum exceeds an adjustable high setpoint (100-140%), the output holds at the high limit. If the sum is less than the adjustable low setpoint (60-140%) than the output holds at the lower limit. The output value then goes to the reactor trip bistable and to the RWP summer. At the reactor trip bistable (programmable) if the indicated reactor power is greater than the trip setpoint (from the summer), a reactor trip signal is produced.

At the RWP summer, the high or low setpoint output is added with the pre-selected RWP offset setpoint. The output is then fed to the RWP programmable bistable. If the comparator goes high, an RWP trip is produced.

In addition to the FTSC, a circuit is added for heat balance correction as shown in Figure 2. The indicated true power as calculated from the heat balance equations in the data logger will be indicated on an added meter. The circuit will correct for non-PPS readouts, megawatt-hour meter, power and flow measurements, and the flux-recorder. It should be noted that this circuit receives an input from the data logger and is not connected with the FTSC nor part of the PPS. This circuit was not reviewed since it is independent of the change for the floating trip setpoint circuit used in the PPS.

Figure 4 shows the test setup to calibrate each nuclear channel. A nuclear channel consists of a detector and one half of a dual linear drawer as shown in Figure 5. Figure 5 shows one channel of the three PPS channels where a coincident logic of two-out-of-three is required to produce a trip.

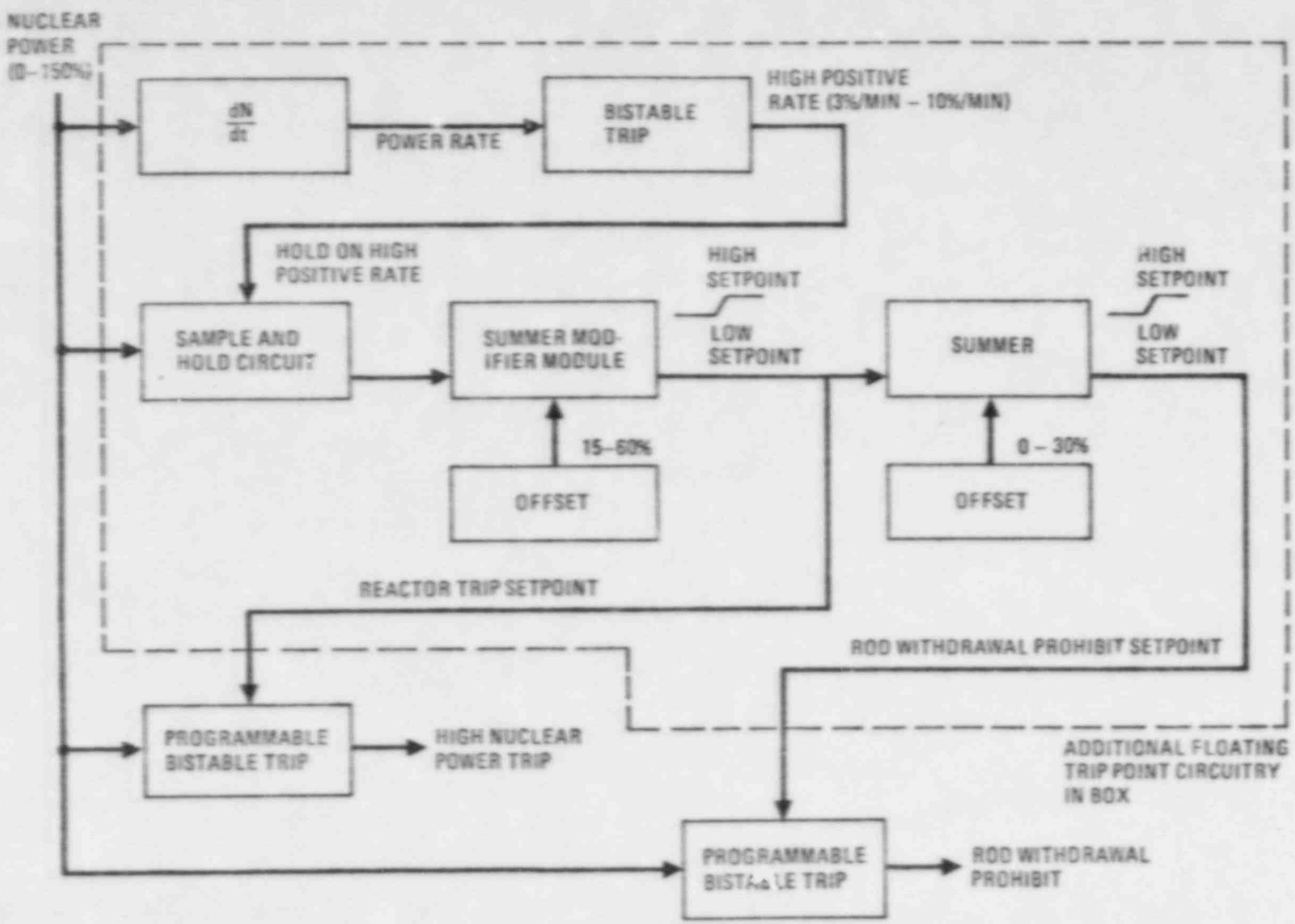


Figure 1. PPS floating trip circuit (typical of six places)

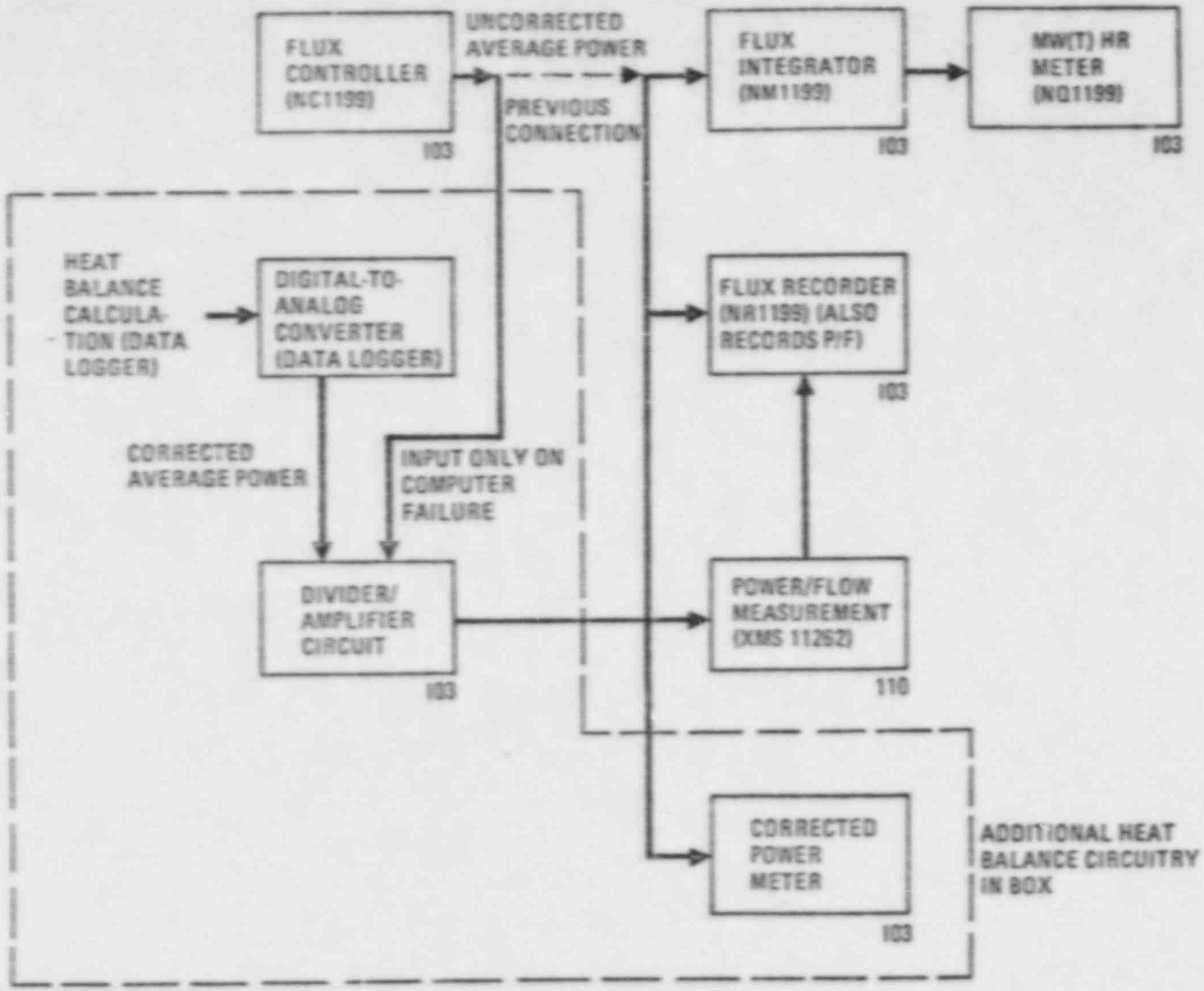


Figure 2. Heat balance calibration

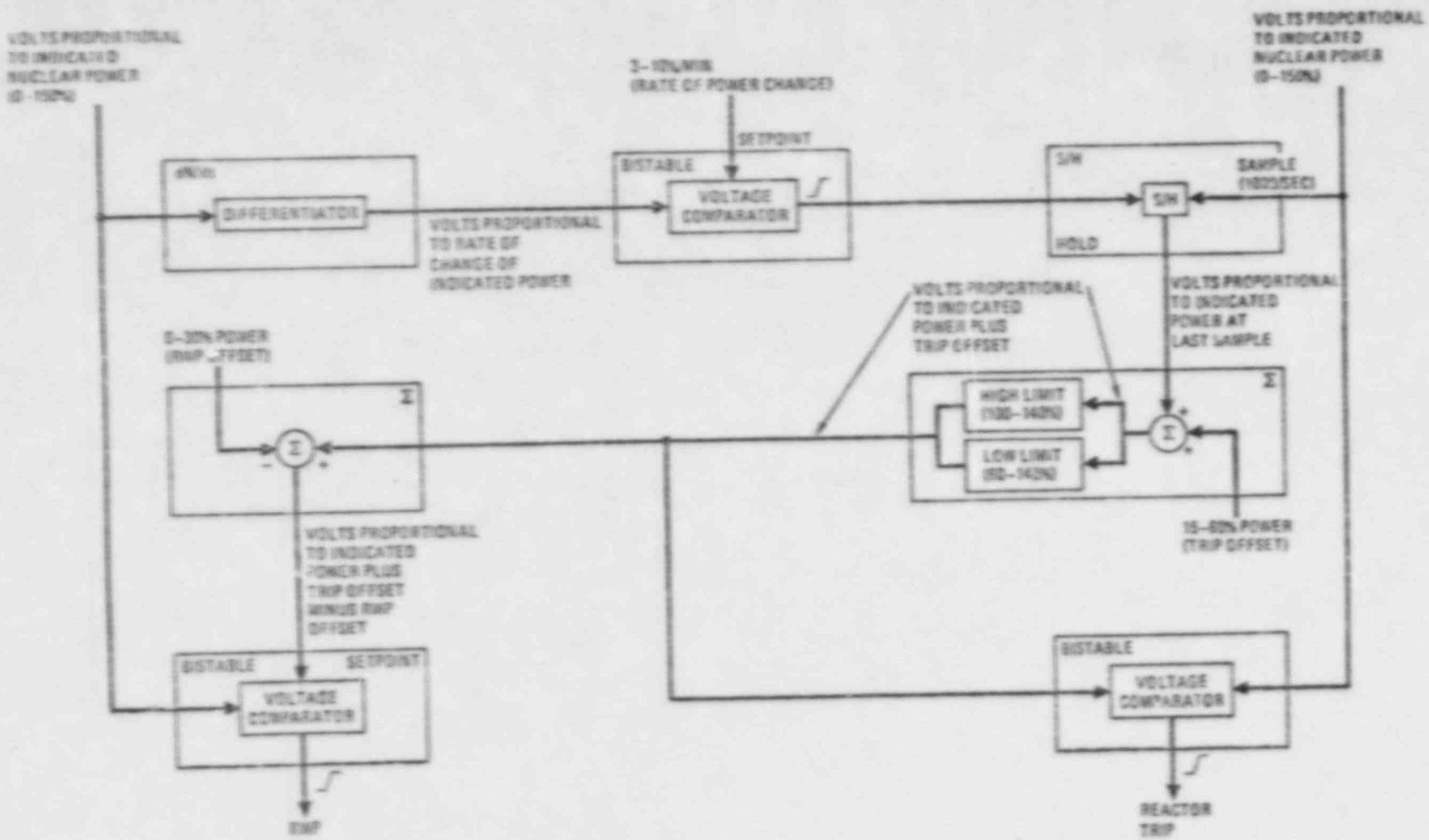


Figure 3. PPS floating trip point circuitry for Fort St. Vrain Unit 1

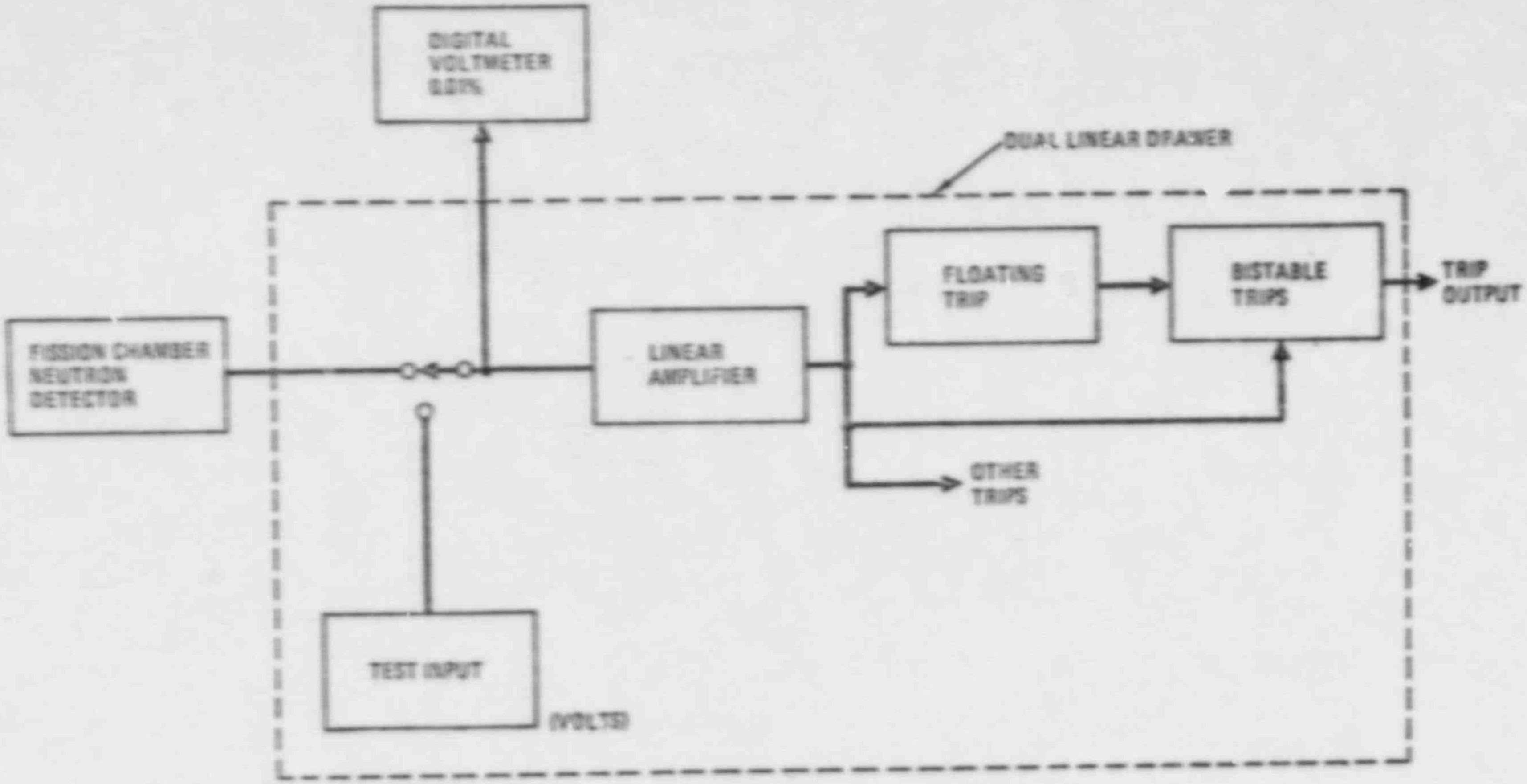


Figure 4. Nuclear channel test setup, block diagram

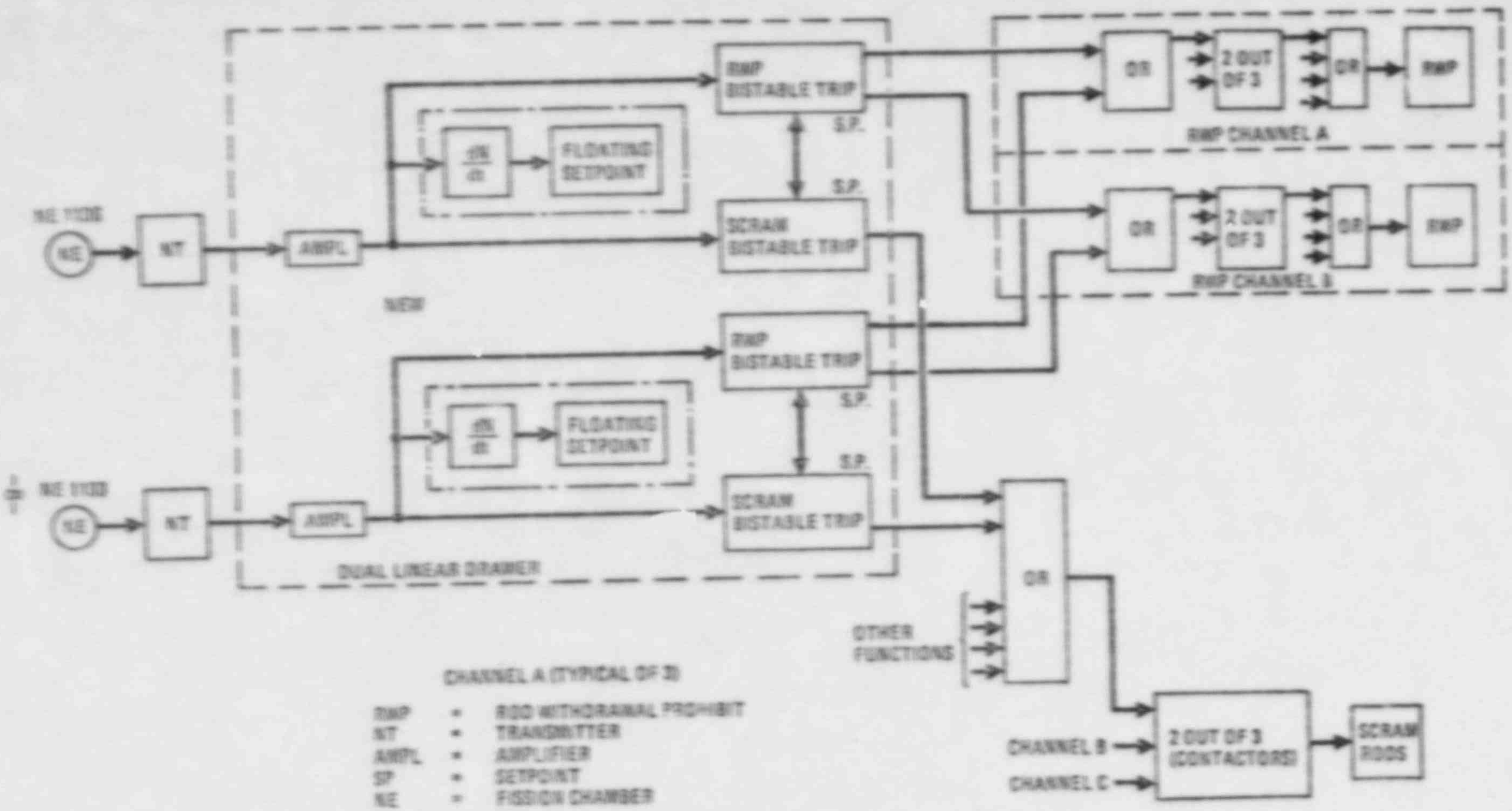


Figure 5. PPS channel configuration

4. EVALUATION

This section presents an evaluation on various aspects of the proposed neutron detector decalibration.

4.1 Circuitry Design Changes

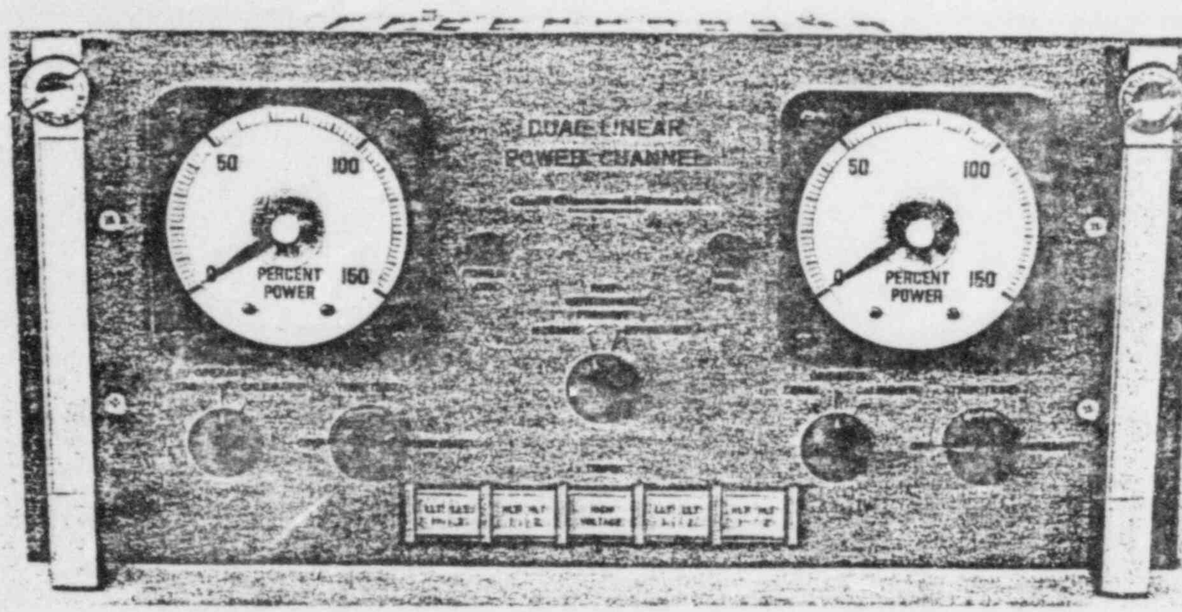
The Fort St. Vrain plant protective system design is based on the 1967 edition of the NRC General Design Criteria [Ref. 5] and IEEE 279-1968 [Ref. 6] as stated in the plant's Final Safety Analysis Report.

The addition of the FTSC to the PPS involves replacing the entire dual linear power range channel drawer and modules with the drawer shown in Figure 6. All the required modules (not shown) for the FTSC are located within the drawer. There are no external modifications to the drawer/modules required.

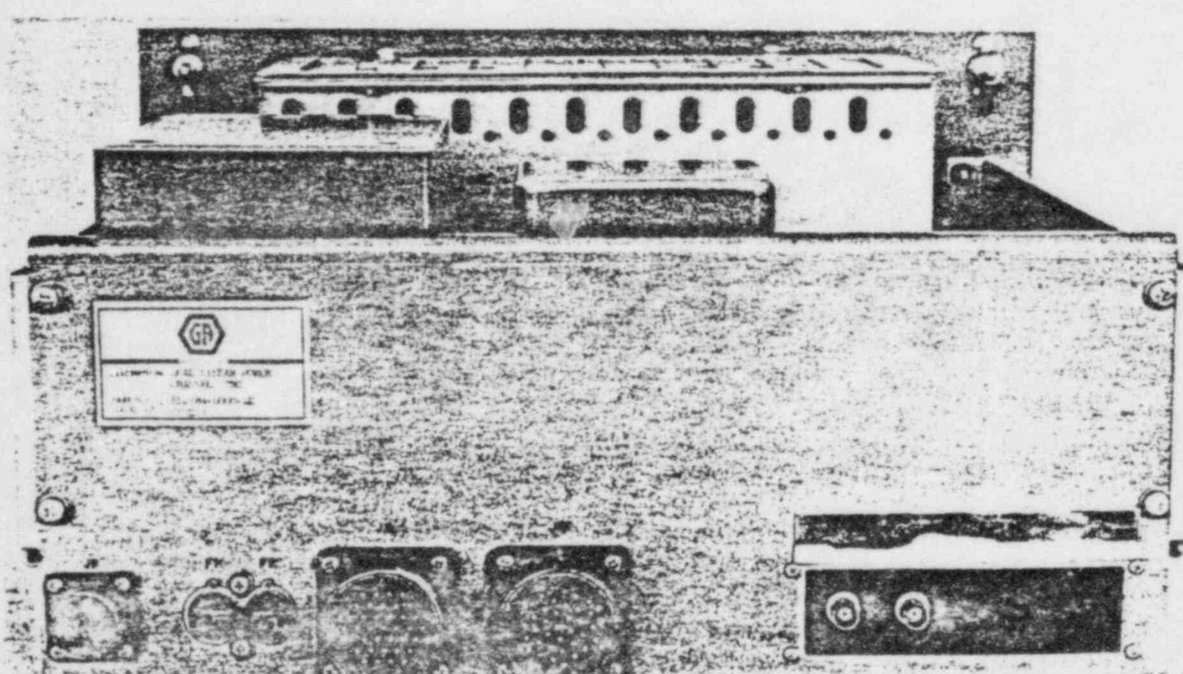
A review of drawings, schematics, and drawer specifications submitted (e.g. dual linear power channel schematic and assembly, floating trip setpoint schematic, bistable trip circuits, period rate circuits, linear amplifier schematic, and operation manual) [Ref. 3] finds that the FTSC does not alter any of the original system design criteria. This includes the criteria of redundancy, overall logic, failure modes, field wiring, arrangement, independence, testability, reliability, or physical separation of the reactor trip circuits [Ref. 4].

A failure modes and effects analysis (FMEA) was performed by General Atomic Company [Ref. 1]. The FMEA analyzed each component of the FTSC with respect to the "Failure Mode," "Channel Effect," and "System Effects." The results of their analysis demonstrated that no single failure of any portion of the FTSC will prevent the PPS from initiating or completing a reactor trip or rod withdrawal prohibit. The addition of the FTSC does not change the failure modes of the original system. The change does add one additional failure mode. This is the failure of a channel to detect a high rate of flux increase. This failure mode will not prevent PPS action since the original system was not rate dependent and the upper trip limit (existing system) is still active to activate a trip.

A review of the FMEA finds that most of the component failures (e.g. shorts, opens, or high output) either causes no detection of high flux rates or causes the trip point not to float (may either go high, low, or zero) at the channel level. These failure modes at the system level result either in a spurious 1-channel RWP and trip or in loss of channel RWP detection with a trip still detectable with the remaining 2 PPS channels. Therefore, component failures within the FTSC will neither prohibit nor adversely affect the PPS from initiating its protective function since the trip limits of the original system are still effective.



FRONT VIEW



REAR VIEW

Figure 6. Dual Linear Channel Drawer

A review of the dual linear drawer schematic finds that the circuit boards and switches are interlocked to provide automatic channel tripping if their performing function is prohibited. The interlock path is accomplished by the +15 Vdc supply where interruption causes the bistable trips and trip relays to de-energize to their tripped condition. For loss of bus voltage or sensor, channel trip is also automatically initiated.

4.2 Calibration Requirements

General Atomic Company recommends the following calibration requirements [Refs. 1 and 2]:

(1) At least one calibration is required during every 24-hr period when operating in low power or power modes.

(2) To prevent or clear RWPs which occur due to inaccurate detector readings, a calibration should be done whenever any channel approaches or reaches an RWP setpoint.

(3) To ensure that the interlock sequence switch (ISS) is switched at the proper power level, the following requirements are made:

a. With the ISS in the startup mode, a calibration is required when heat-balance power is between 2% and 4% of rated power. The methods to determine heat-balance power level are given in Technical Specification Surveillance Procedure 5.4.1.1.4 c-D.

b. When increasing power with the ISS in the low power mode, a calibration is required when heat-balance power is between about 24% and about 28% of rated power.

c. When decreasing power with the ISS in the low power mode, a calibration is required when heat-balance power drops below about 35% of rated power.

(4) Whenever the operator has reason to believe that one or more detectors are giving anomalous readings, a calibration should be performed.

(5) Whenever individual detectors differ by more than 10%, the proper functioning should be verified.

(6) Add the following items to the existing FSV calibration procedure, S.R.5.4.1.1.4c-D:

Control rod bank partially inserted ______________
Position (inches withdrawn) ______________
Regular rod position (inches withdrawn) ______________

(7) Calibrate detectors prior to the withdrawal of rod group 3C (for cycle 2).

The uncertainties analysis used to establish the above calibration requirements considered instrumentation uncertainties, calorimetric uncertainties (heat balance equations), and conservative uncertainties in the analysis used to establish the required trip setpoints. Taking these uncertainties into account, a margin of safety (% of rated power) larger than the total uncertainties (%) was established to ensure that a trip occurs below the Technical Specification values.

margin = (required trip setpoint) - (programmed trip setpoint)

This margin was conservatively fixed at 7% to 28% of rated power.

A review of the above General Atomic Company calibration requirements finds that they will add assurance that the FTSC will adjust the PPS trip setpoints as a function of indicated power within a margin of safety to ensure the TS limits are not exceeded.

4.3 Technical Specifications and Test Provisions

The limiting conditions for operation (LCOs) are in Table 4.4-1 and Table 4.4-3 and the surveillance requirements are shown in Table 5.4-1 and Table 5.4-3 of the FSV Technical Specifications. The required trip setpoints, minimal number of operable channels, minimum degree of redundancy, and permissible bypass conditions are listed. A change is required in the test provisions to include the manufacturer's detailed test procedures for the new circuitry.

5. CONCLUSIONS

Based on the information submitted by the Public Service Company of Colorado, on the neutron detector decalibration, at the Fort St. Vrain Nuclear Generating Station, it is concluded that:

(1) The proposed circuitry changes for adjusting the PPS trip setpoints (reactor and RWP) as a function of indicated power, meets the NRC design requirements as given in Section 2 above and as stated in the plant's Final Safety Analysis Report.

(2) The licensee make necessary changes in operating procedures to follow General Atomic Company's recommendations on the calibration requirements for the proposed circuitry changes.

(3) The test provisions in the plant procedures be changed to accomodate the manufacturer's detailed test procedures for the added circuitry.

Accordingly, I recommend that the NRC staff accept the neutron detector decalibration circuitry change to adjust the PPS trip setpoints as a function of indicated power.

REFERENCES

(1) Public Service Company of Colorado (J. K. Fuller) to the NRC (W. P. Gammill), dated January 11, 1979.

(2) Public Service Company of Colorado (J. K. Fuller) to the NRC (S. A. Varga), dated November 29, 1979.

(3) Public Service Company of Colorado (H. L. Brey) to the NRC (J. T. Collins), dated May 16, 1983.

(4) Public Service Company of Colorado (H. L. Brey) to the NRC (E. H. Johnson), dated November 30, 1983.

(5) General Design Criterion 7, 12, 15, 19-26, and 39, "AEC General Design Criteria for Nuclear Power Plant Construction Permits," 1967 edition.

(6) IEEE Standard: "Criteria for Protection Systems for Nuclear Power Generating Stations," IEEE 279-1968.

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