ANL-76-34 ANL-76-34

257
5-24-78

FINAL SAFETY ANALYSIS ADDENDA TO HAZARDS SUMMARY REPORT, EXPERIMENTAL BREEDER REACTOR II (EBR-II): UPGRADING OF PLANT PROTECTION SYSTEM, VOLUME I

MASTER

APPLIED TECHNOLOGY

Any further distribution by any holder of this document or of the data therein to third parties representing foreign interests, foreign governments, foreign companies and foreign subsidiaries or foreign divisions of U. S. companies should be coordinated with the Director, Division of Reactor Development and Demonstration, U. S. Energy Research and Development Administration.

U of C-AUA-USERDA

ARGONNE NATIONAL LABORATORY, ARGONNE, ILLINOIS

Prepared for the U. S. ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION under Contract W-31-109-Eng-38

The facilities of Argonne National Laboratory are owned by the United States Government. Under the terms of a contract (W-31-109-Eng-38) between the U. S. Energy Research and Development Administration, Argonne Universities Association and The University of Chicago, the University employs the staff and operates the Laboratory in accordance with policies and programs formulated, approved and reviewed by the Association.

Printed in the United States of America
Available from
U. S. Energy Research and Development Administration
Technical Information Center
P. O. Box 62
Oak Ridge, Tennessee 37830
Price: Printed Copy $9.00

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF FIGURES

LIST OF TABLES

LIST OF TABLES

FINAL SAFETY ANALYSIS ADDENDA TO HAZARDS SUMMARY REPORT, EXPERIMENTAL BREEDER REACTOR II (EBR-II): UPGRADING OF PLANT PROTECTION SYSTEM. VOLUME I

Compiled by

J. I. Sackett and N. L. Gale

ABSTRACT

This report is a compilation of the formal Final Safety Analysis Addenda (FSAA's) to the EBR-II Hazard Summary Report and Addendum that have been prepared in support of certain modifications to the reactor-shutdown-system portion of the EBR-II plant protection system. Each major section is an edited version of the original FSAA for a particular modification and provides a description of the pre- and post-modification system, the rationale for the modification, and required supporting safety analysis.

1. INTRODUCTION

The Experimental Breeder Reactor is an unmoderated, heterogeneous, sodium-cooled fast breeder reactor operated by the Argonne National Laboratory at the Idaho National Engineering Laboratory (INEL), formerly the National Reactor Testing Station. EBR-II is a pool reactor, with the core, primary pumps, and intermediate heat exchanger submerged in a tank of sodium. The core consists of (1) driver subassemblies of uranium metal fuel elements, (2) some experimental subassemblies containing fuels and materials being irradiated, and (3) control and safety subassemblies. The nominal power output is 62.5 MWt (20 MWe), with a neutron flux at the core center of about 2.5×10^{15} cm^2/s.

The Final Safety Analysis Addenda (FSAA) to the EBR-II Hazard Summary Report[1,2] described in this report are issued to fulfill requirements of ERDA 0540 (Ref. 3) for safety documentation. The modifications for upgrading the Plant Protection System (PPS) have been approved by ERDA and implemented. These FSAA's have been designated as final, in accordance with Sec. 0411 of ERDA 0540,[3] because they are based on final design information.

Publication of FSAA's as ANL reports is required after ERDA concurrence, generally in the sequence of modification.

The modifications pertain to different parts of the reactor shutdown system.

2. EBR-II PLANT MODIFICATION NO. WAF-796: UPGRADING OF REACTOR SHUTDOWN SYSTEM

V. N. Thompson

2.1 Summary

To eliminate the possiblity of failure of the PPS arising from a failure within the reactor shutdown string itself, a second shutdown string, system B, is incorporated in the shutdown system. The output of system B and that of existing system A are connected in one-out-of-two logic to interrupt power to the control-rod latches to start a reactor trip. The systems are electrically isolated and physically separated from each other to prevent loss of protection due to an internal random failure or a credible single event. Each system can be checked for proper operation without bypassing any trip contacts. The tests must be conducted with the reactor shut down.

The input and the control-power relays of system B are of the same type as those used in system A. The instrument channels in this modification continue to function as before the modification.

System B includes only PPS functions that have been shown by analyses to be essential for plant protection: protective functions associated with reactivity insertion, loss of coolant flow, subassembly outlet temperature, and earthquake Should other plant-protection subsystems be required to prevent reactor operating conditions from exceeding safety limits, contacts from the identified instrument channels will be added to system B.

System B does not directly perform any interlock or control functions, but a contact in the control-rod-down interlock causes system A to perform these functions. This feature is provided because an instrument channel that has contacts in both systems A and B could function or malfunction in such a way that only system B would receive the shutdown signal from a single channel or group of channels connected in coincidence. Redundancy in the interlock and control functions is not considered essential, and the added complexity that would be introduced into these circuits if system B were connected to perform these functions independently is considered undesirable.

This modification was completed in April 1975.

2.2 System before Modification

The design of the original (A) shutdown system is shown in Fig. 35 of Ref. 2. In this modification, a redundant (B) shutdown system was installed.

2.3 System after Modification

Shutdown system B consists of relay and switch contacts connected in series and series-parallel to form an electrical circuit through which power is supplied to the control-power relays of system B. If this circuit is interrupted, a reactor trip results.

A schematic diagram for system B is shown in Fig. 1. Devices and subsystems included are described below.

2.3.1 Supervisory, Test, and Manual-shutdown Switches

This group includes four switches:

a. Reactor-control-power Key Switch 1PK. This switch, on the reactor control console, controls electrical power to both shutdown systems.

b. String-test Key Switch ITKS. This switch, on the control console, has contacts in both shutdown systems and provides means to interrupt each system separately for testing. A contact is also provided to bypass the CP-B contact in the control-rod-down interlock circuit (see Fig. 2) to prevent the CP-A relays from opening during testing of the system B contacts in the control-rod-latch circuits. Contacts on this switch also provide capability for checking control-rod drop times using either shutdown system.

c. Safety-rod Trip Switch 3MPB. This switch on the control console also opens systems A and B.

d. Control-rod Trip Switch 1MPB. This switch on the control console also opens systems A and B.

2.3.2 Earthquake-protection Subsystem

The schematic diagram for the earthquake-protection subsystem of system B is shown in Fig. 3. Included are three channels arranged in two-out-of-three trip logic. The protective channels formed for system B have relay contacts at each detector. The earthquake-protection subsystem for system A remains the same as before PM-WAF-796.

2.3.3 Reactivity-related Trips

The reactivity-related trips include the period and the high-power-level trips. Figure 4 is a schematic diagram for the period-protection subsystem. The period trip is bypassed above a preset power level. The minimum bypass level (25 MW) is specified by EBR-II technical specification limits. Figures 5 and 6 are schematic diagrams for the period-bypass subsystem and the high-power-level protective subsystem. Each of these subsystems include three protective channels arranged in two-out-of-three logic.

The new wide-range nuclear instrumentation was installed under Plant Modification No. WAF-753 (see Sec. 3). This instrumentation uses a dual-bistable approach to provide separate bistables for each shutdown system.

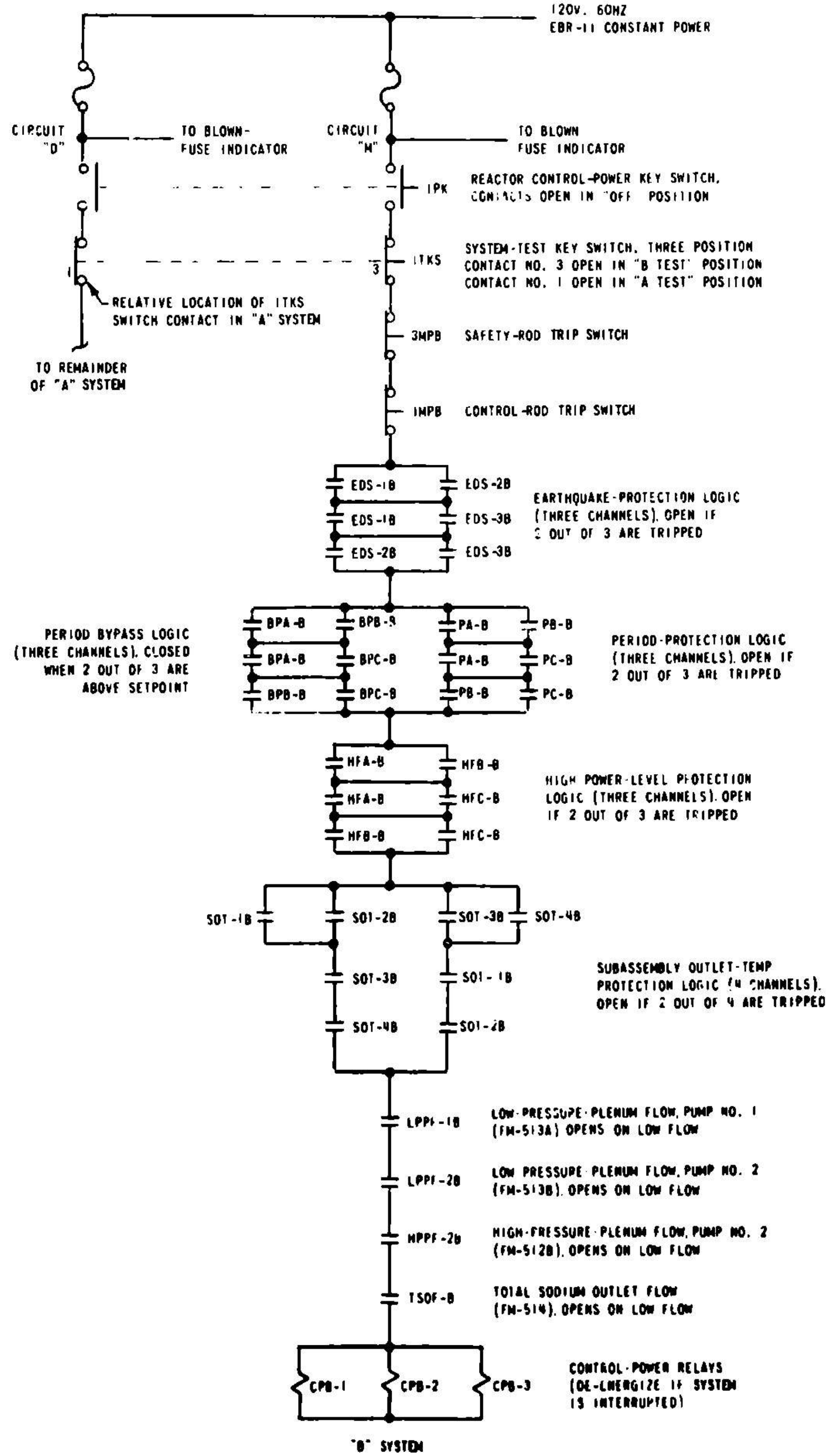


Fig. 1. Schematic Diagram for Shutdown System B

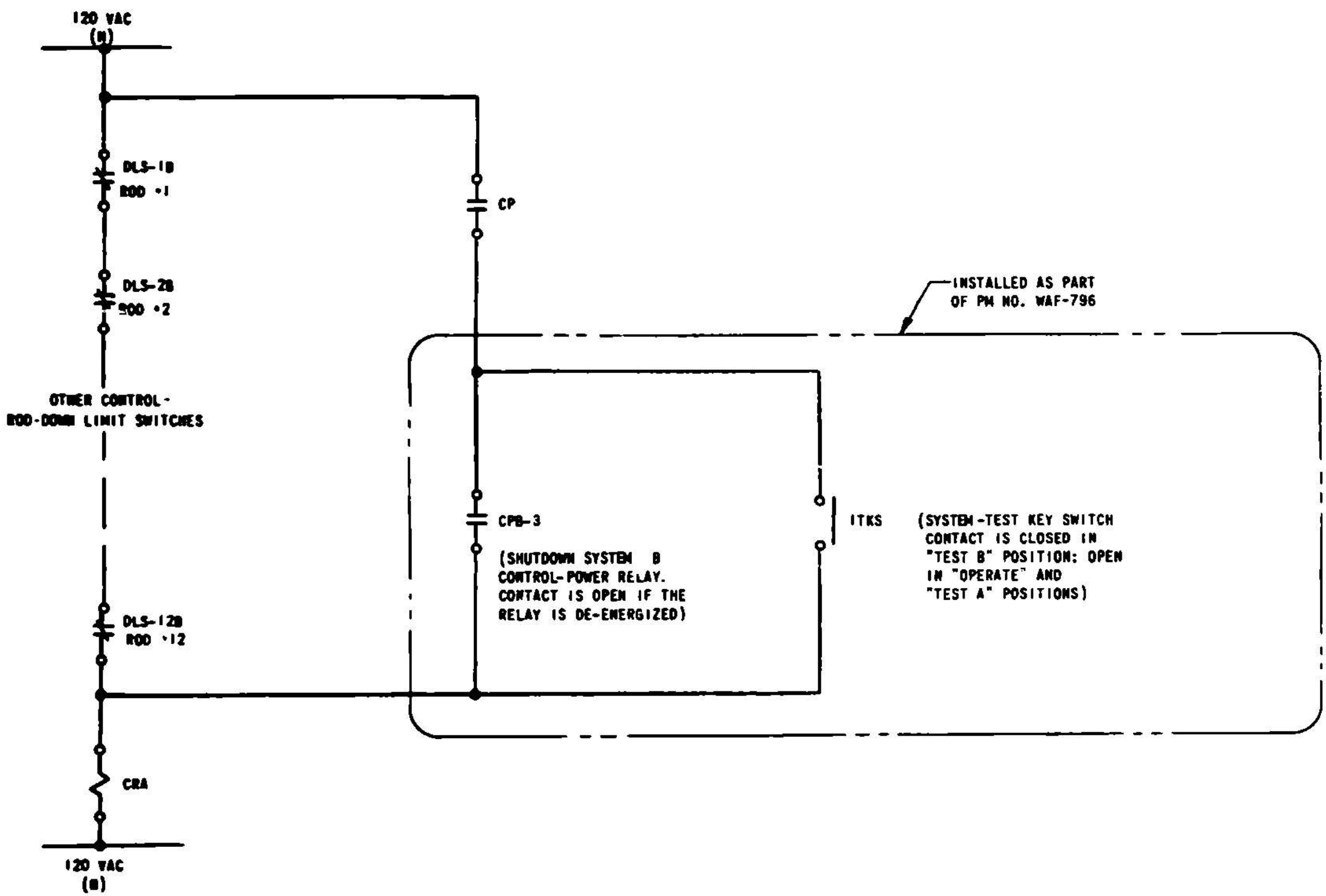


Fig. 2. Schematic Diagram of Modified Control-rods-down Auxiliary-relay (CRA) Circuit

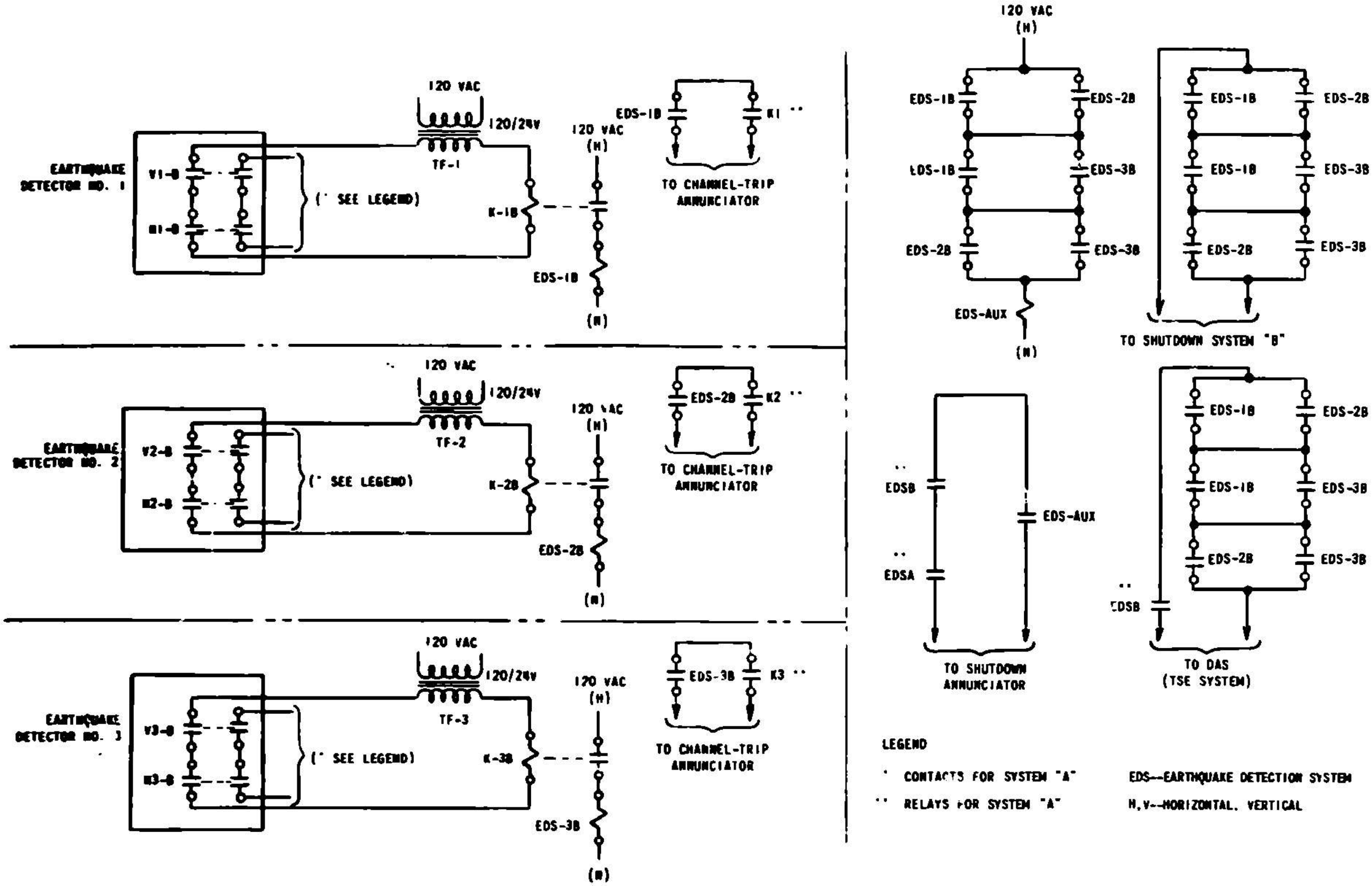


Fig. 3. Schematic Diagram for Earthquake-protection Subsystem of Shutdown System B

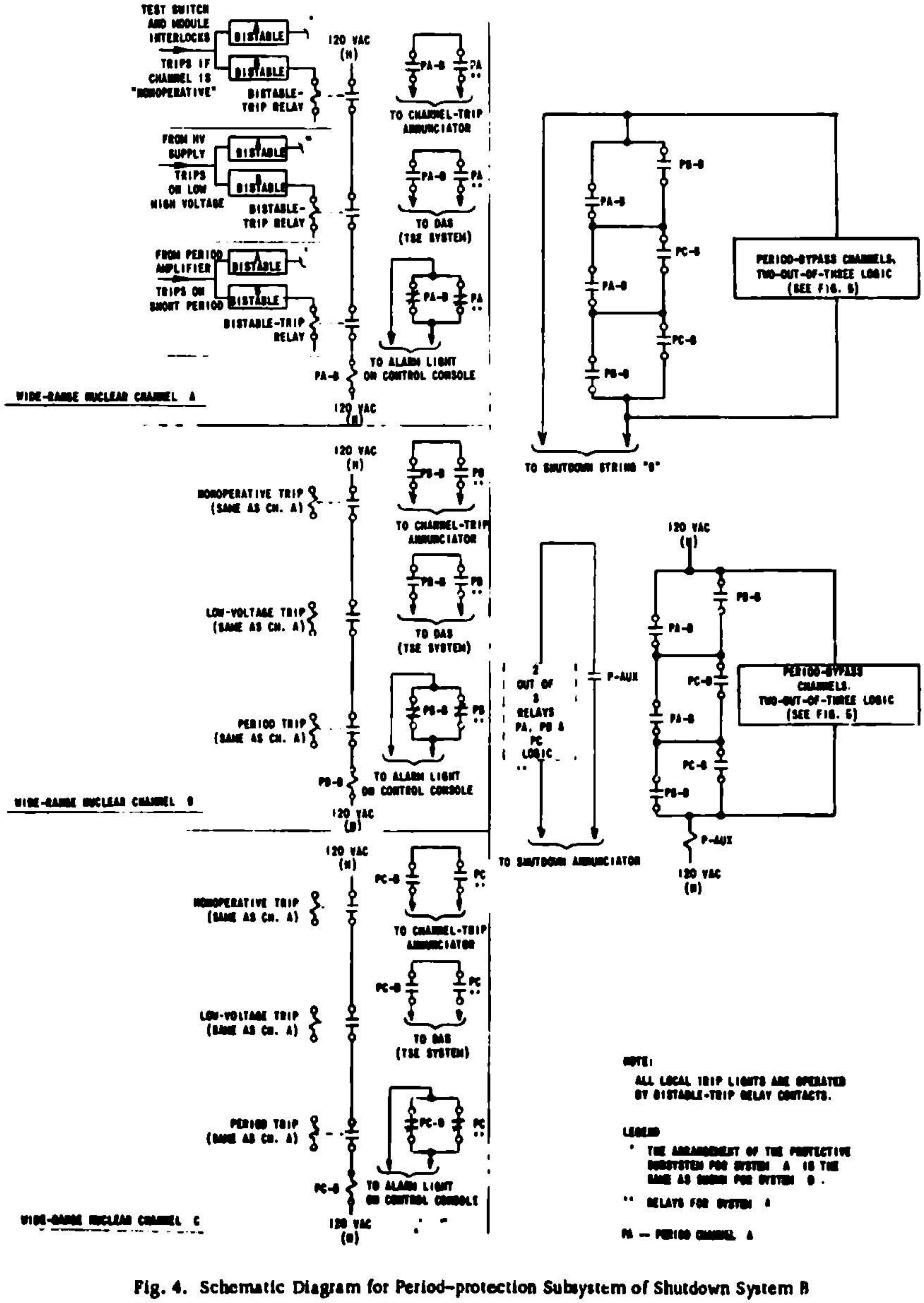


Fig. 4. Schematic Diagram for Period-protection Subsystem of Shutdown System B

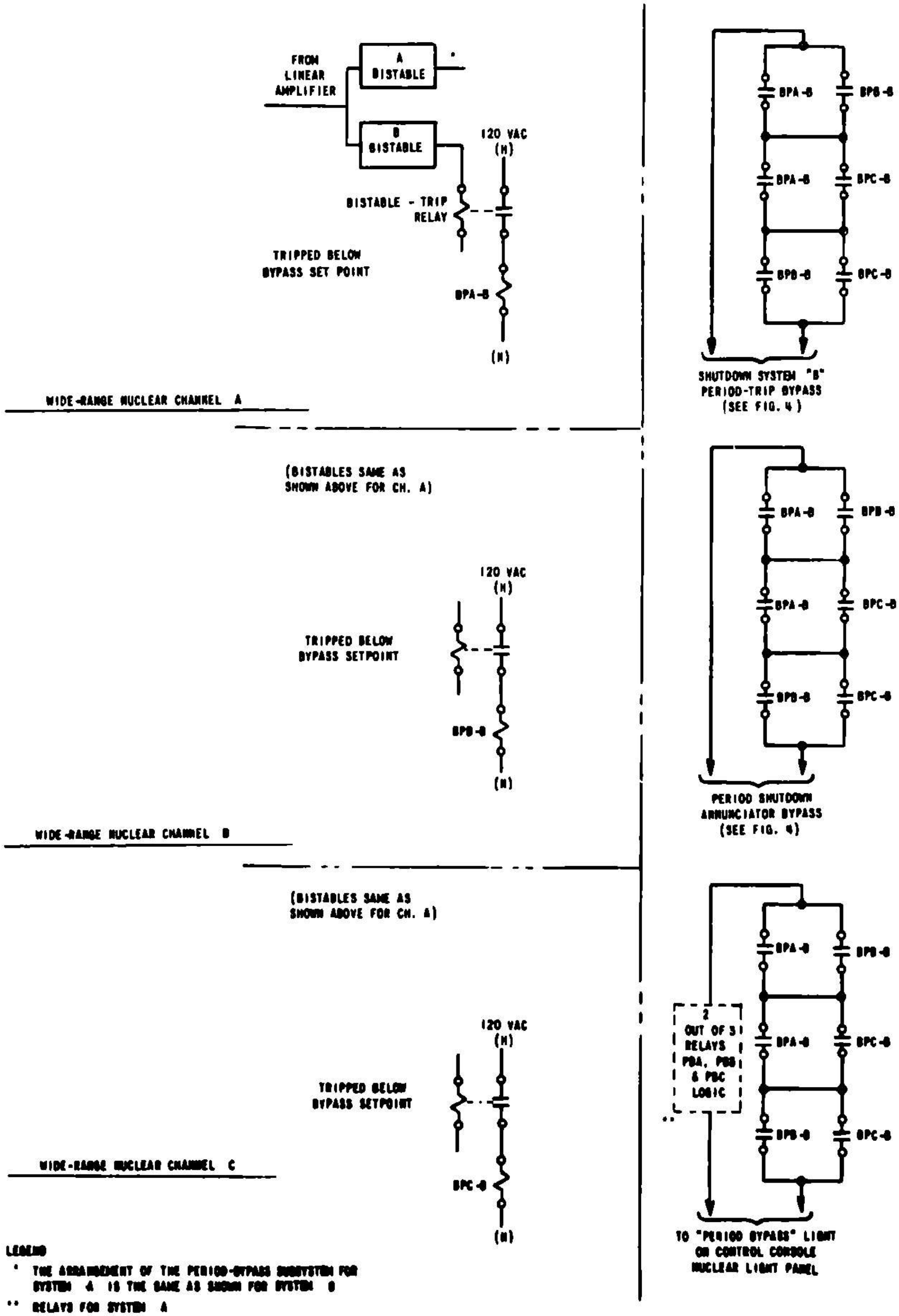


Fig. 5. Schematic Diagram for Period-trip Bypass of Shutdown System B

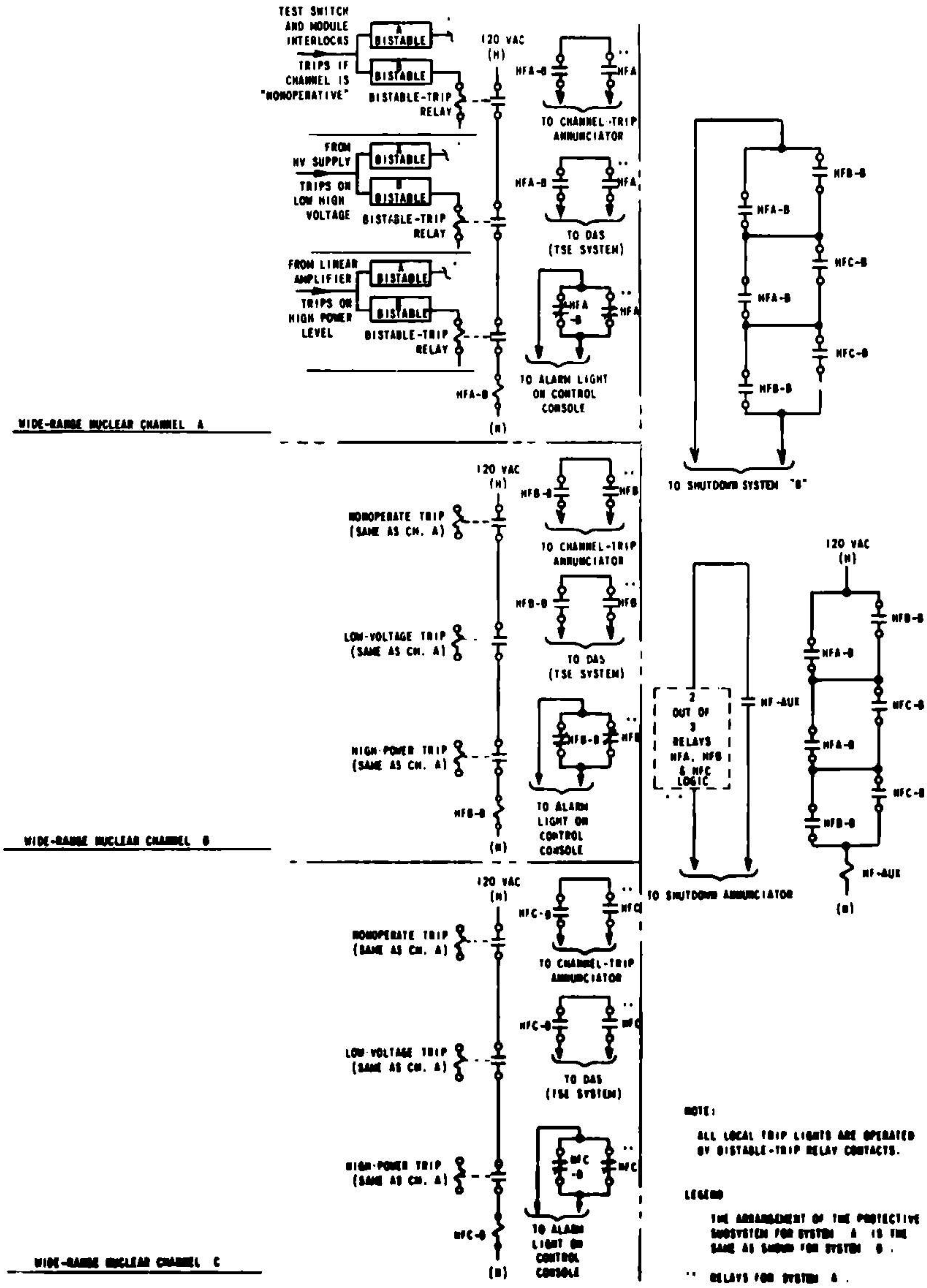


Fig. 6. Schematic Diagram for High-power-level-protection Subsystem of Shutdown System B

2.3.4 Trips for Subassembly Outlet Temperature

Figure 7 is a typical schematic for each of the four channels for subassembly outlet temperature. The trips are arranged in two-out-of-four logic in system B as shown in Fig. 1. The channels were upgraded under PM No. 443 (see Sec. 4) to meet the requirements for separation and isolation of RDT C16-1T[4] insofar as practicable. These channels have a dual-bistable arrangement, which provides a separate bistable for each shutdown system.

2.3.5 Trips Related to Loss of Primary Flow

This group includes four protective channels for primary-sodium flow. Figures 8-10 are schematic diagrams for the flow channels, each of which operates as a one-out-of-one trip in the shutdown system.

The protective channels for loss of primary flow were upgraded under PM No. 443 (see Sec. 4) to meet the requirements for separation and isolation of RDT C16-1T.[4] These channels use a dual-bistable arrangement, which provides a separate bistable for each shutdown system.

2.3.6 Location of Components of System B

Bistables, and the bistable-trip relays, serving system B, are located with the corresponding channel instrumentation. All the manual switches, described in Sec. 2.3.1 are on the reactor control console. The input and control-power relays of the system are in a group of four cabinets in the cable routing room (CRR). (See Fig. 11.)

2.3.7 Control-rod-latch Circuit

The system B output (control-power) relay contacts are incorporated in the circuit for the control-rod electromagnetic latch (see Fig. 12). Two control-power relays, CPB-1 and -2, are used. Relay CPB-1 serves rods 1-6, and relay CPB-2 serves rods 7-12. Separate contacts are provided for each rod.

2.3.8 Modification of Shutdown System A

Shutdown system A is modified to incorporate contacts from the string test key switch 1TKS described in Sec. 2.3.1. These contacts are connected in system A immediately after the contacts of switch 1PK, as shown in Fig. 1, to coincide with the arrangement for system B.

2.3.9 Alarm Indications

The alarm indications in the control room, related to the operation of an instrument channel connected to perform a trip function, were not changed by WAF-796. Trips from either system A or B provide an alarm that is functionally identified independent of the affected system.

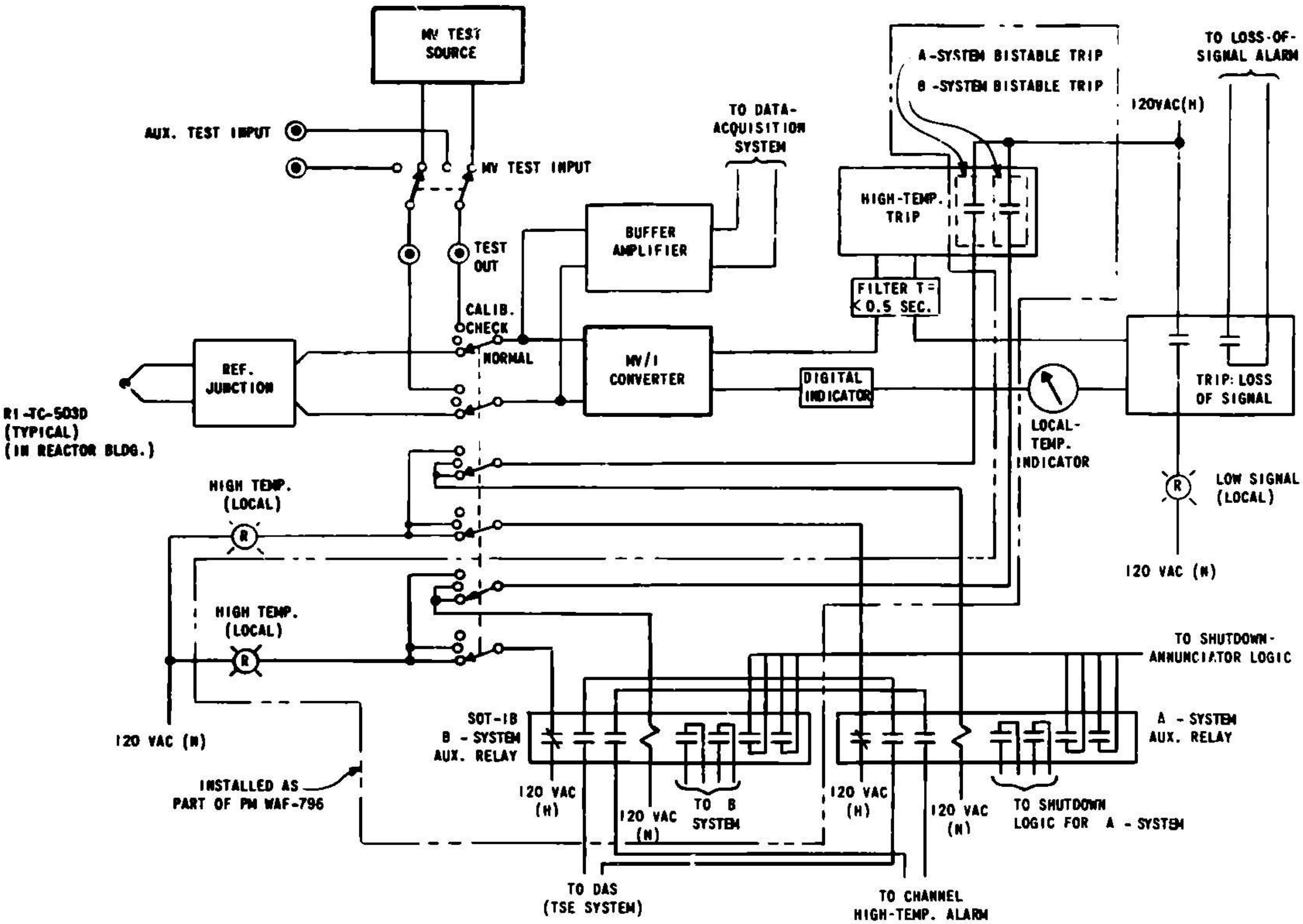


Fig. 7. Schematic Drawing of Channels for Subassembly Outlet Temperature

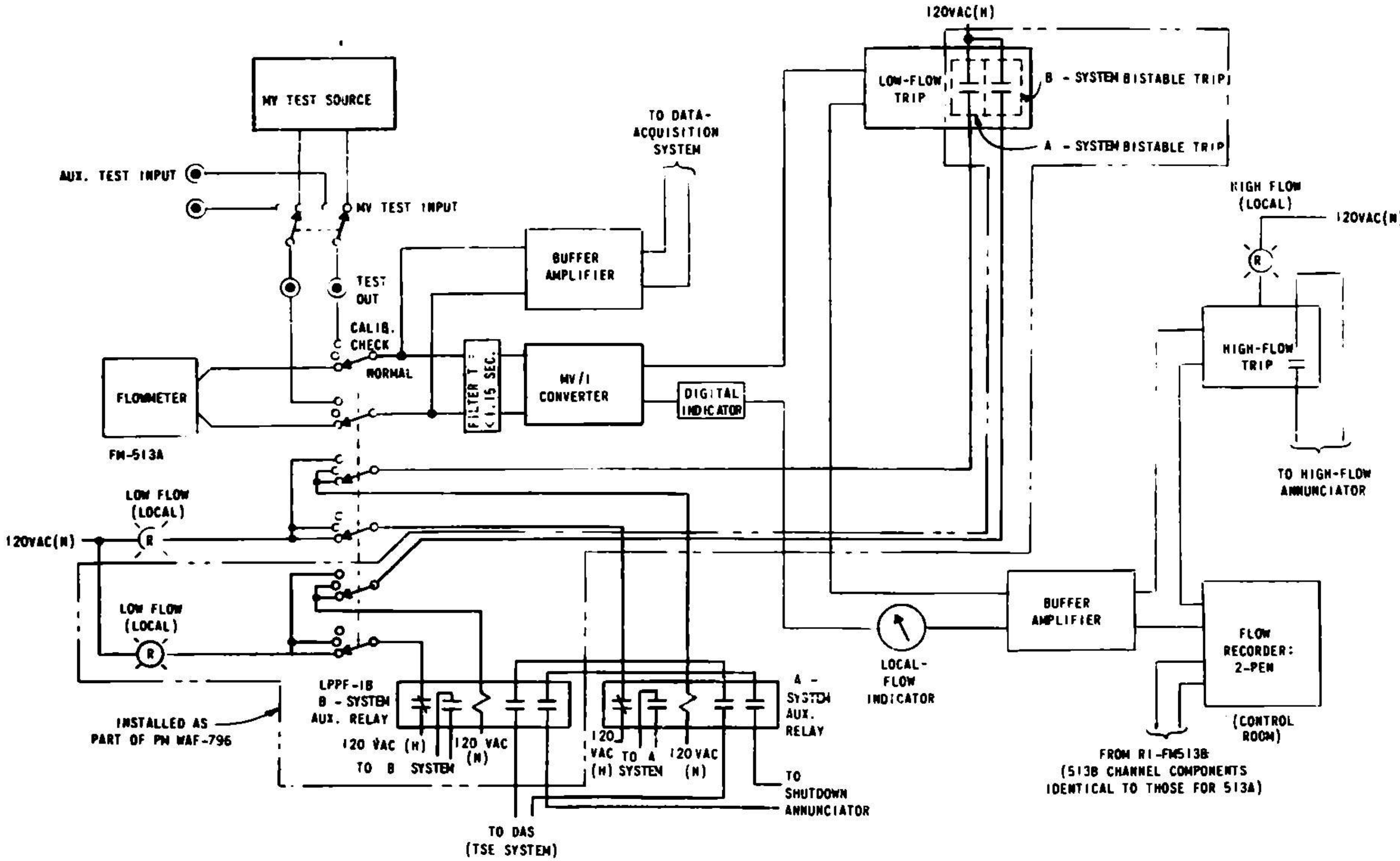


Fig. 8. Schematic Diagram of Channels for Low-pressure-plenum Flow

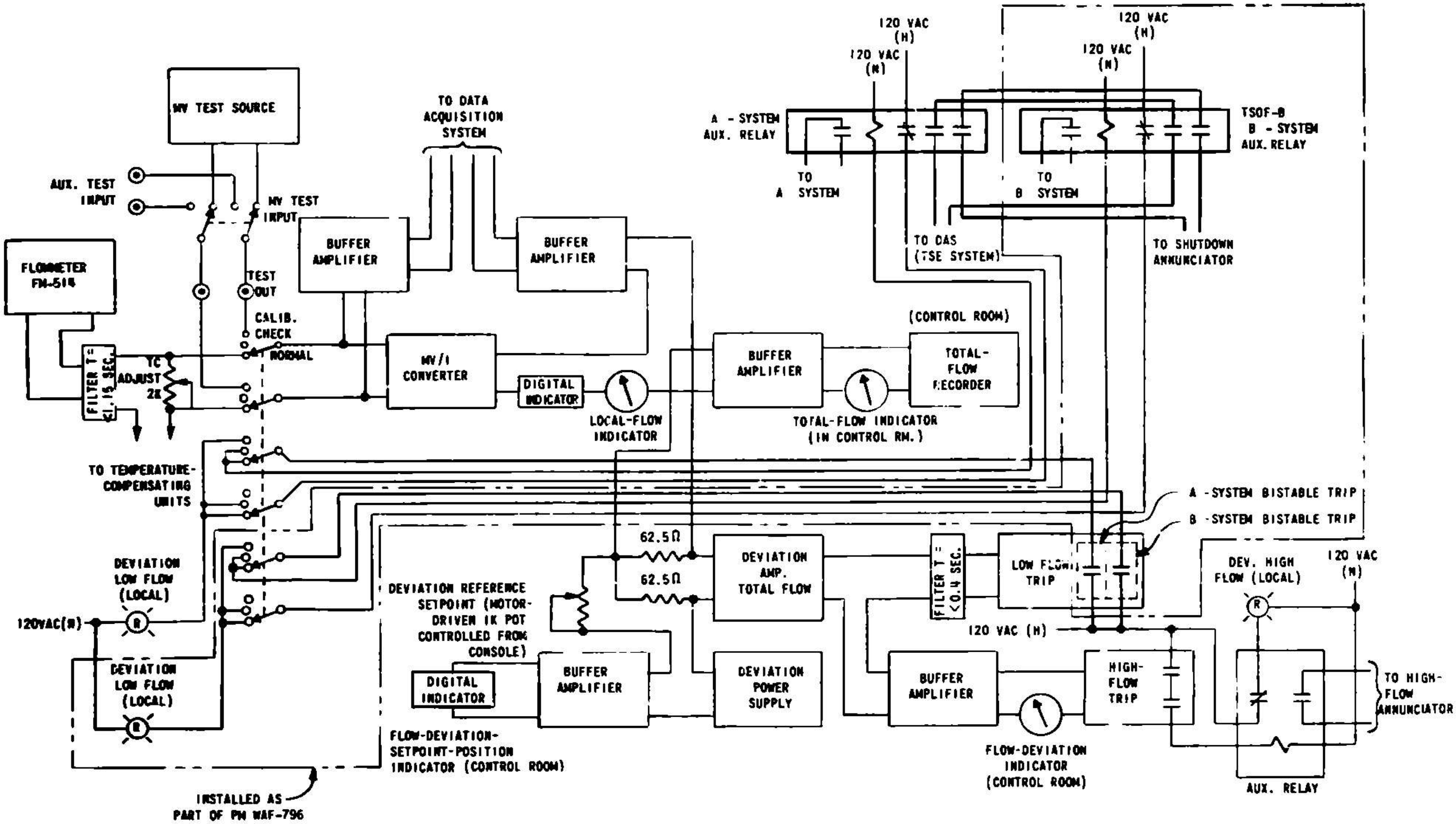


Fig. 9. Schematic Diagram of Channels for Total Reactor Flow

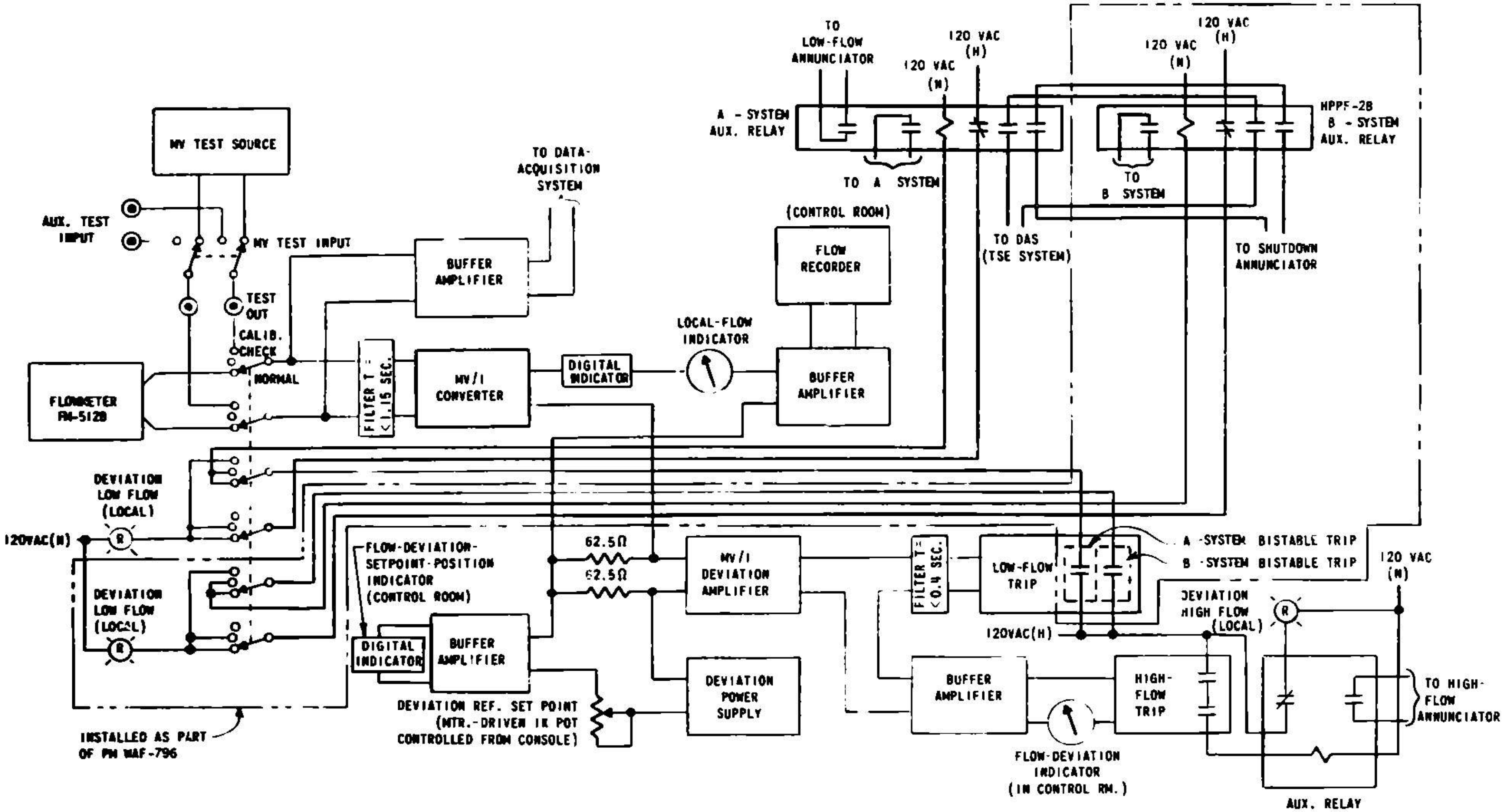
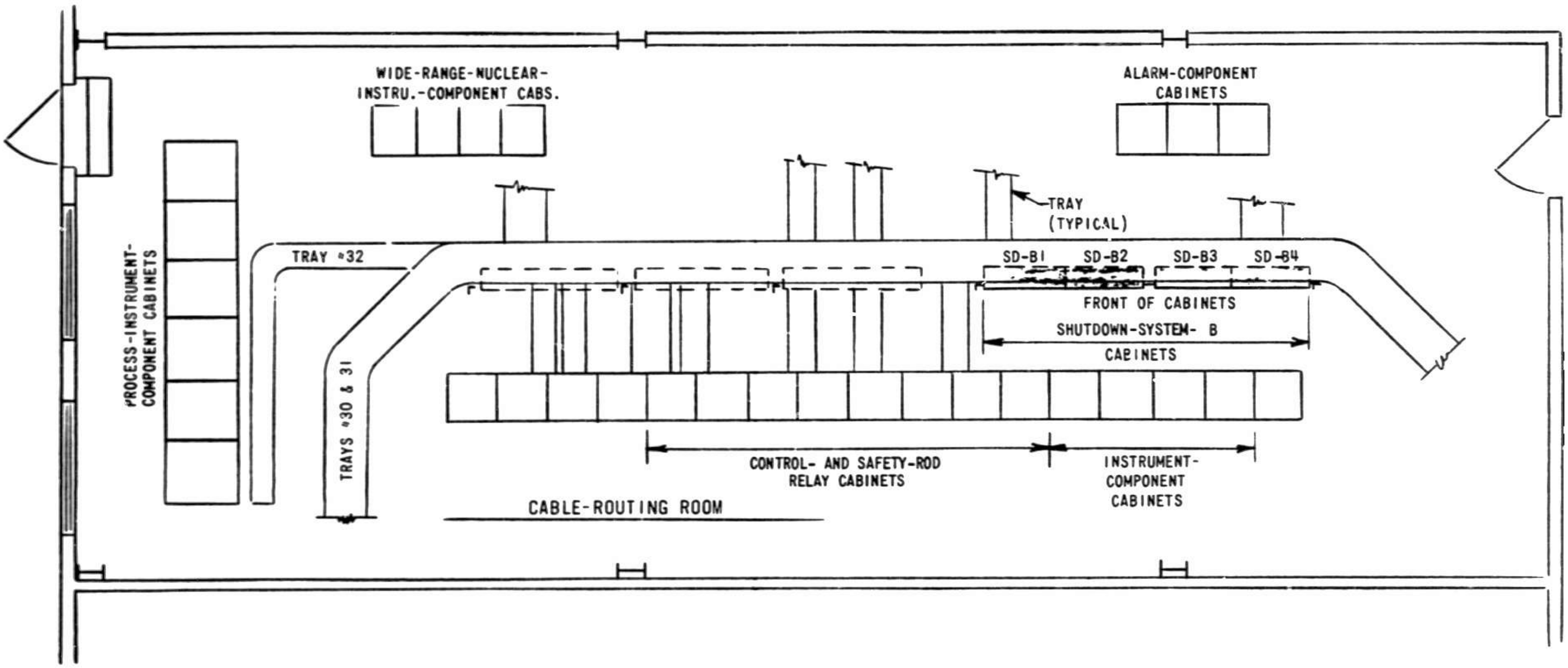


Fig. 10. Schematic Diagram of Channels for Pump No. 2 High–pressure–plenum Flow



SCALE: [illegible]" = 1'-0"

Fig. 11. Plan View of Cable-routing Room, Showing Location of Cabinets of Shutdown System B. Conversion factors: 1 in. = 25.4 mm; 1 ft = 0.3048 m.

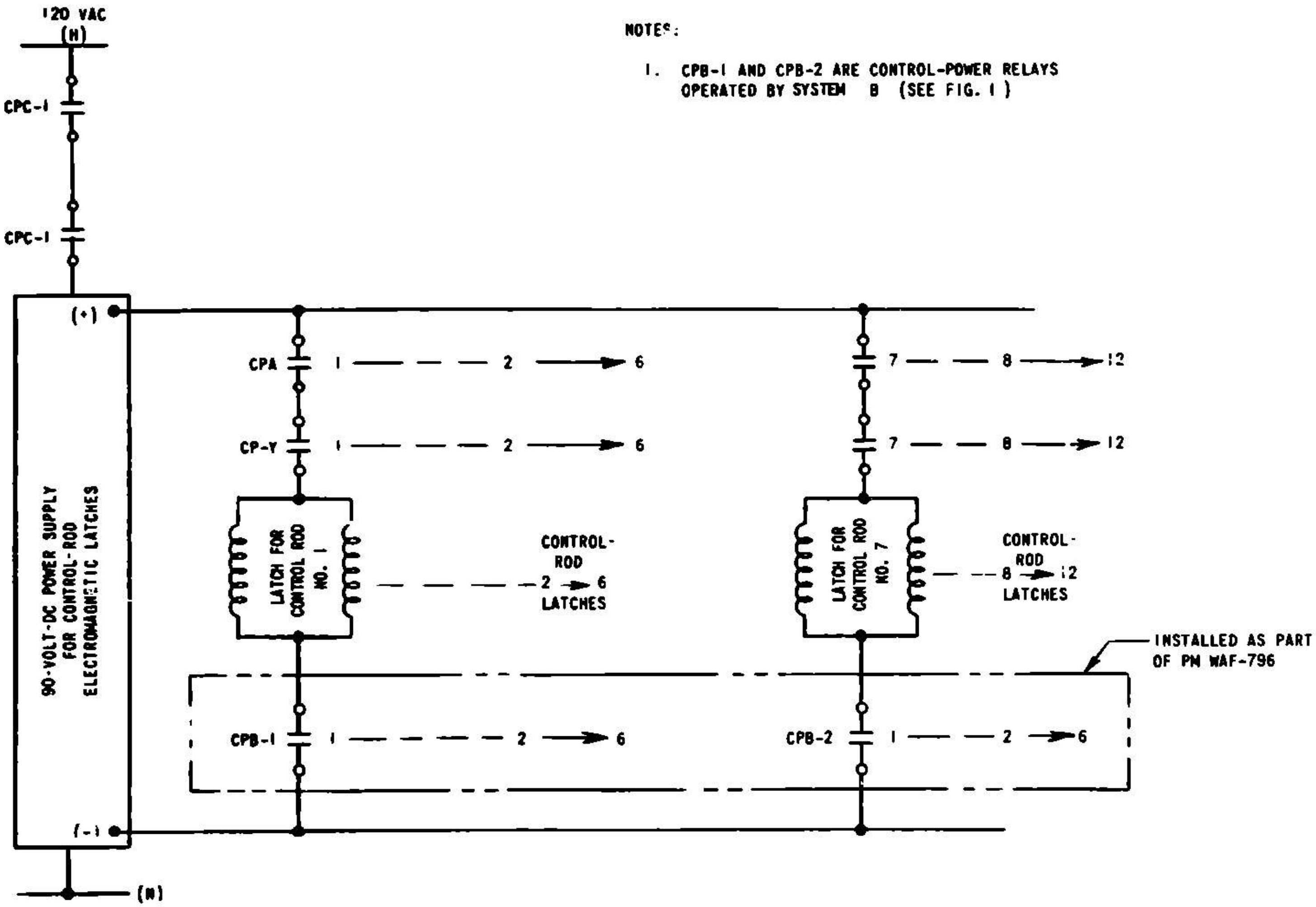


Fig. 12. Schematic Diagram of Control-rod Electromagnetic Latch

Normally, both systems function at about the same time, and the operator's response is independent of which system started the trip. Independent alarms show when each system is deenergized so that the operator can easily determine if one system has malfunctioned. The operation of each shutdown system is similarly indicated in the data-acquisition system (DAS) time-sequence-of-events (TSE) record.

2.3.10 Control and Interlock Functions

The CP-B relays do not directly perform any interlock or control functions, nor does any interlock or control function provide an input to the circuit of system B. When the CP-B relays are deenergized, a bypass contact in series with the CP-A contact in the CRA circuit (see Fig. 2) opens, and the CRA circuit opens if the control rods are not all in the down position. This in turn opens the CP-A relays, which in turn perform the same functions as would have been performed if system A had started the shutdown.

2.3.11 Electrical Power

Power supplied to system B from the EBR-II constant-power system is 120 V, 60 Hz.

2.4 Justification for Modification

Plant Modification No. WAF-796 eliminates the possibility of a single short circuit within the shutdown string preventing the protective action of reactor shutdown. This modification is specifically directed to upgrading the EBR-II PPS by bringing the EBR-II reactor shutdown system into conformance with RDT Standard C16-1T.

2.5 Applicable Standard

This modification is governed by the requirements of RDT C16-1T;[4] it does not require any variances to these requirements.

2.6 Safety Analysis

2.6.1 Redundancy and Separation

The modified protective logic includes two shutdown systems arranged in one-out-of-two logic to trip. To guard against an internal random failure or a credible single event resulting in a simultaneous failure of both systems, these systems are separated electrically and physically. Electrical isolation between the systems is achieved by separate bistables, bistable-trip relays, and string-input relays for each system. Housing the input and control-power relays for each system in separate cabinets provides physical separation.

Elements common to both systems are switches for reactor-control power, system test, and manual shutdown (two switches). All are in the reactor control console. The systems in this area are separated by using separate contact blocks on the switches for each system and running the conductors for each system in separate cables. To cope with the possibility of a common-mode failure in this area, these switches are connected in the end of the system nearest the supply line. Consequently, no loss of protection from any protective subsystem can result from a common-mode failure in this part of the circuit.

In system B, redundant instrument channels are separated by locating the system-trip relays for a given protective subsystem in separate cabinets. Interconnecting cables between these relays and the instrumentation racks are in separate conduits. The four protective channels for primary-sodium flow, even though arranged in one-out-of-four logic in system B, are considered redundant for providing protection for a whole-core, loss-of-flow accident. Therefore, the string-input relays for these channels are also separated by installation in separate cabinets.

Separate conduits isolate wiring of the PPS from other wiring. This includes the wiring between cabinets for system B and the connecting cables between the string-input relays and the instrumentation racks. In the system-B cabinets, PPS and non-PPS wiring are in separate bundles. To further minimize possible interaction between the wiring, the cabinets for system B are restricted solely to components of this system.

In summary, the modified protective logic described above is considered to conform with the requirements for redundancy and separation of RDT C16-1T.

2.6.2 Testing Capability

Operation of system B can be fully tested with the reactor shut down. Switch 1TKS provides means to deenergize each shutdown system separately for testing and to check the control-rod drop time with either system.

2.7 Revisions to Practices and Documentation

2.7.1 Operations

The instrument channels in this modification perform the same functions as before the modification. EBR-II technical-specification limits have been revised to include reference to all functions contained in system B.

No changes in required operator actions during normal operation or after a trip are required as a result of this modification.

Operating instructions were revised before operation of the reactor with this modification installed to (1) include a discussion of system B,

(2) point out that two shutdown systems (A and B) exist, and (3) incorporate specific instructions for routine checkout of system B in procedures for interlock checkout.

2.7.2 Administrative Controls

Administrative controls required for safe operation of the plant are not affected by this modification.

2.8 Conclusion

Installation of WAF-796 does not adversely affect safety of the EBR-II reactor plant during normal operations or under any accident conditions discussed in Ref. 1 or previous addenda. This modification improved the reliability of the PPS.

3. EBR-II PLANT MODIFICATION NO. WAF-753: UPGRADING OF REACTIVITY PROTECTION

L. J. Christensen, R. N. Curran, E. M. Dean, and W. K. Lehto

3.1 Summary

The reactivity-protection subsystem of the EBR-II reactor shutdown system must provide protection under the accident conditions for which power level or period protection was stated in Refs. 1 and 2. The wide-range instrument channels meet this requirement.[5] Additional requirements for the reactivity protection subsystem, related to the protection of the plant from damage, are in RDT C16-1T.[4] The plant-damage limitations are the controlling requirements.

The three wide-range channels serve trip functions, during reactor operation, equivalent to those previously provided by nuclear channels 1-6 and 9-11. (See Fig. 35 of Ref 2.) The period-trip function is automatically bypassed when a power of 30 MWt is reached, because analysis shows that the period trips are not effective above this power and that power-level trips provide adequate protective margin.[6]

The wide-range channels provide protection from reactivity-insertion accidents during refueling. These channels are connected in a two-out-of-three coincidence arrangment instead of the one-out-of-three arrangement shown on Fig. 35 of Ref. 2. The two-out-of-three coincidence arrangement is permitted by the technical-specification limits and provides adequate system reliability.[5]

The modification does not introduce any new safety hazards and protects against any accident protected against by the original nuclear-instrumentation system.

This modification was completed in May 1975.

3.2 System before Modification

Performance characteristics of the pre- and postmodification nuclear channels are compared in Table I.

TABLE I. Performance Characteristics of Original and Wide-range Nuclear Channels

Characteristic	Original Channels	Wide-range Channels
Log Count Rate (LCR)	Channels 1-3	Channels A-C
Manufacturer	EG&G	Gulf (LCR subsystem)
Source Count Rate	Scaler output after pulse-height discrimination	LCR scaler output after pulse-height discrimination
Range	Six decades logarithmic; $1\text{-}10^6$ cps	Six decades logarithmic; $1\text{-}2 \times 10^5$ cps

TABLE I (Contd.)

Characteristic	Original Channels	Wide-range Channels
Log Count Rate (LCR)	Channels 1-3	Channels A-C
Accuracy	1 cps ± 5% 10 cps ± 2% 10^2 cps ± 1% 10^3 cps ± 1% 10^4 cps ± 1% 10^5 cps ± 1% 10^6 cps ± 2%	±1% at 10^2, 10^4, and 2×10^5 cps measured at the 10-decade output
Trip Circuits		
Count Rate/High	Fuel-handling trip set at 1500 cps; trip logic is one out of three.	Fuel-handling trip set at 1500 cps; trip logic is two out of three.
Count Rate/Low	Contact opens when level drops below 10 cps; trip logic is one out of three.	Contact opens when level drops below 10 cps; trip logic is one out of three.
Trouble	Provides a light when the LCR chassis cover is removed.	Provides annunciation if either cover, any circuit board, or any output MS connector is removed; or if switches are not in operate positions.
Trip Specifications	Each high, low, and period trip is derived from two parallel independent trip comparators.	Each high, low, and period trip is derived from two parallel independent trip comparators.
	Each channel provides two sets of output. Contacts arranged for: Closed--operate Open--trip (de-energize relay)	Each channel provides two sets of output. Contacts arranged for: Closed--operate Open--trip (de-energize relay)
Response	10-90% of LCR output to a factor of two-step increase.	0-63% in 10-decade output to step increase given.

Original Channels:

Level, cps	Response Time, s
1-2	10.00
10-20	10.00
$1\text{-}2 \times 10^2$	1.00
$1\text{-}2 \times 10^3$	0.10
$1\text{-}2 \times 10^4$	0.01
$1\text{-}2 \times 10^5$	0.01

Wide-range Channels:

Level LCR, cps	Response Time, s
4-40	7.00
10-100	2.00
100-1000	0.50
10^3-10^4	0.05
10^4-10^5	0.03
MSV[a], nv	
10^4-10^6	0.04
10^6-10^7	0.02
10^7-10^{10}	0.015

Characteristic	Original Channels	Wide-range Channels
Drift	<2% of full-scale output for 100 h.	<1% of equivalent linear full scale in 100 h.
Calibration	Two points at 60 Hz and 100 kHz ± 0.1%. Variable dc-voltage adjustment to set trip points.	Six points, 10, 10, and 2×10^5 cps for LCR and 5 μs; 10 kHz 0.3, 0.9, 10 V for AMS. Variable dc-voltage adjustment to set trip point.

TABLE I (Contd.)

Characteristic	Original Channels	Wide-range Channels
Period Circuit	Channels 1-3	Channels A-C
		Period signal is obtained from combined LCR and AMS signal; at low power, only LCR contributes.
Range	-100 to +10 s	-100 to +10 s
Period	Contact opens when period is less than 25 s. Trip logic is one out of three in fuel handling and two out of three in reactor operate.	Contact opens when period is less than 25 s. Trip logic is two out of three in the common circuit.
Response	Time constant is nonlinear and approaches 20 s when period level is from -100 to +100 s.	Time constant (0-63%) to a full-scale ramp input to the period differentiator is 7.8 s.
Drift	Less than 1% of full scale in 100 h of continuous operating at design operating conditions.	<1% of equivalent linear full scale in 100 h.
Calibration	One point at +10-s period. A variable ramp adjustment sets the trip setpoint.	Two points, +10- and +40-s periods. A variable dc-voltage adjustment sets the trip setpoint.
Log N	Channels 4, 5, and 6	Channels A, B, and C
Manufacturer	Gulf	Gulf (AMS subsystem)
Range	10^{-11}-10^{-3} A (~1 W-125 MW)	2 x 10^{4}-2 x 10^{10} nv (~125 W-125 MW)
Response Time	Output reaches 63% of final value in 3.2 s for a ramp input corresponding to a 20-s period.	Same period response in counting and AMS modes.
Period Calibration	One fixed point at +10 s and one variable to set the trip setpoint.	Two points, +10 and +40 s; also variable dc voltage to set the trip setpoint.
Trips	Period trip and level bypass trips	Two trips: period and LCR-AMS interlock
Linear Power Range	Channels 9, 10, and 11	Channels A, B, and C
Manufacturer	EG&G	Gulf (linear-power-range subsystem)
Range	~150 W-150 MW by linear range changing	~150 W-150 MW by linear range changing
Accuracy	<0.5%	<0.5% as measured at recorder output at worst-case variations of temperature and line voltage
Response	Response time varies depending on range. Bandwidth varies from 0.3 Hz at 10^{-11} A to 10 kHz at 10^{-4} and above.	Recorder and trip signals reach 63% of final value in less than 0.5 ms

TABLE I (Contd.)

Characteristic	Original Channels	Wide-range Channels
Linear Power Range	Channels 9, 10, and 11	Channels A, B, and C
Trips	One high-level trip function	One high-level trip function
High-voltage Power Supplies	All Channels	Channels A, B, and C
Manufacturer	Power Design	Gulf (unit housed in linear power drawer)
Output	1-2000 V dc (depending on type of chamber) at 0-10 mA	-750 V dc at 5 mA
Regulation	<0.0025% plus 2 mV for ±0% line variation and 100% change in load	<1% for zero to 5 mA change in load current
Trip	Opens contacts on low detector high voltage	Opens contacts on detector high voltage at -700 V dc

[a]Mean square voltage.

The flux-deviation-system channel has been eliminated as part of this modification. This circuit and the "automatic" flux trips were designed as part of the control system for operating the reactor at a constant ratio of power to flow, although EBR-II has never been operated in this constant mode. Operation has been with a constant flow from startup to full power; this operation has required the flux-deviation circuits to be referenced to 100% power. Thus, the automatic flux trips performed the same function as the manual ones.

Because both the automatic and manual trip comparators derived their input signal from the same instrument, they provided redundancy only in comparator and trip-logic circuitry. The unnecessary complexity of the flux-deviation circuitry reduced reliability and increased the possibility of spurious reactor trips without significantly increasing the reliability of the instruments in performing their intended plant-protection function.

3.3 System after Modification

The nuclear-instrumentation system for the EBR-II reactor must safely, accurately, and reliably measure the neutron flux, proportional to reactor power, over the complete range from reactor shutdown to operation at above full power. Assuring this safe and reliable operation requires that significant changes in flux will be detected during all phases of shutdown maintenance, fuel handling, reactor startup, approach to power, and full-power operation. To cover this broad flux range, a wide-range (W-R) nuclear-instrumentation system has been installed. The system consists of three identical, redundant, and independent channels, each of which can cover the required neutron-flux range. A W-R channel has three circuits, each measuring a different neutron-flux range: (a) log count rate (LCR), (b) average magnitude squared (AMS), and (c) linear power level (LPR, dc range).

Installation of the W-R channels has modified the original EBR-II nuclear system, with these channels replacing startup channels 1-3, the intermediate channels 4-6, and the power-level channels 9-11. The modification has not affected operation of channel 7 or 7A.

This modification was completed in May 1975.

3.3.1 System Arrangement

The complete nuclear instrumentation system (see Fig. 13) consists of the three W-R channels plus the two linear power-level channels (7 and 7A). Each of the three independent and redundant wide-range channels consists of a fission chamber, a preamplifier, a log (10-decade) power drawer, a linear power drawer, meters, recorders, trip relays, and annunciators. Each channel covers the flux range from shutdown to full power and provides all the trip functions, readouts, and annunciations provided by the original channels.

3.3.2 Readouts

The W-R system has eight readout instruments in the reactor control room. On these instruments, 21 signals can be indicated by channel switching. Three scaler-timers on the nuclear panel provide information on pulse count rate. An audible monitor has a switch to select the channel desired. The reactor period is indicated on a console meter, and the period signal is recorded on a recorder on the nuclear panel. Both the meter and recorder have a switch to select the desired channel. The LCR and AMS signals are recorded on the console recorder, which has two switches to select the desired signals. The 10-decade signal is recorded on another console recorder, which has a switch to select the desired channel. Each of three signals for linear power level is recorded on a separate recorder on the nuclear panel.

LCR pulse and level signals are at the fuel-handling console. A recorder with a selector switch records the selected LCR signal. Two scaler-timers provide information on pulse count rate. The two signals are selected by connection of the desired signal cables.

LCR, 10-decade, period, linear level, and high-voltage signals are indicated locally on the meters mounted on the log and linear W-R chassis.

A dual scaler for count rate and an LCR level recorder are in the fuel-handling console.

The W-R signals are also connected to the DAS system for data collection and alarm when required.

3.3.3 Shutdown Circuit

Figure 14 shows the arrangement of the W-R channels in the shutdown circuit. The shutdown circuit common to both the fuel-handling and

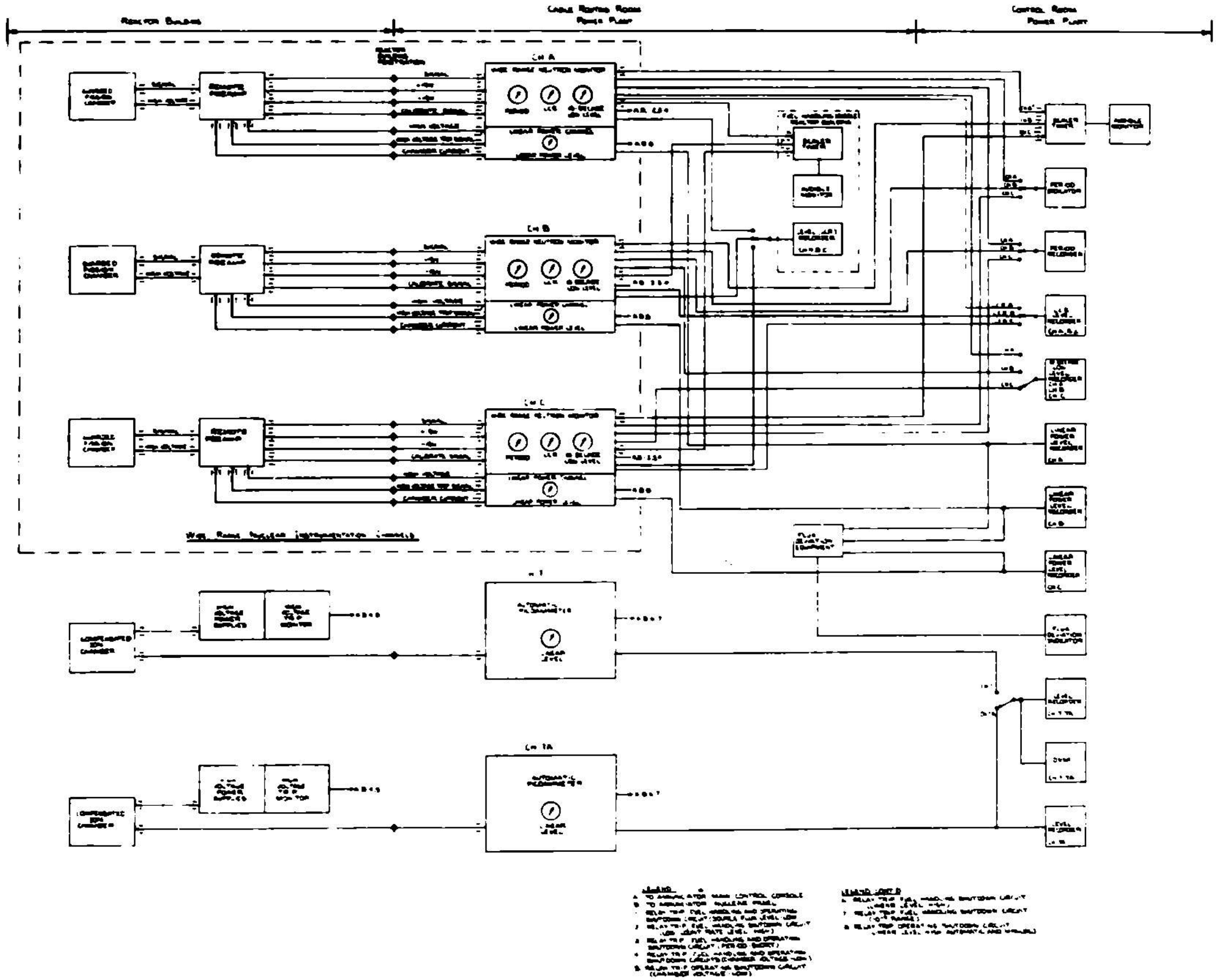


Fig. 13. Schematic Drawing of Wide-range Channels. ANL Neg. No. 103-A9564.

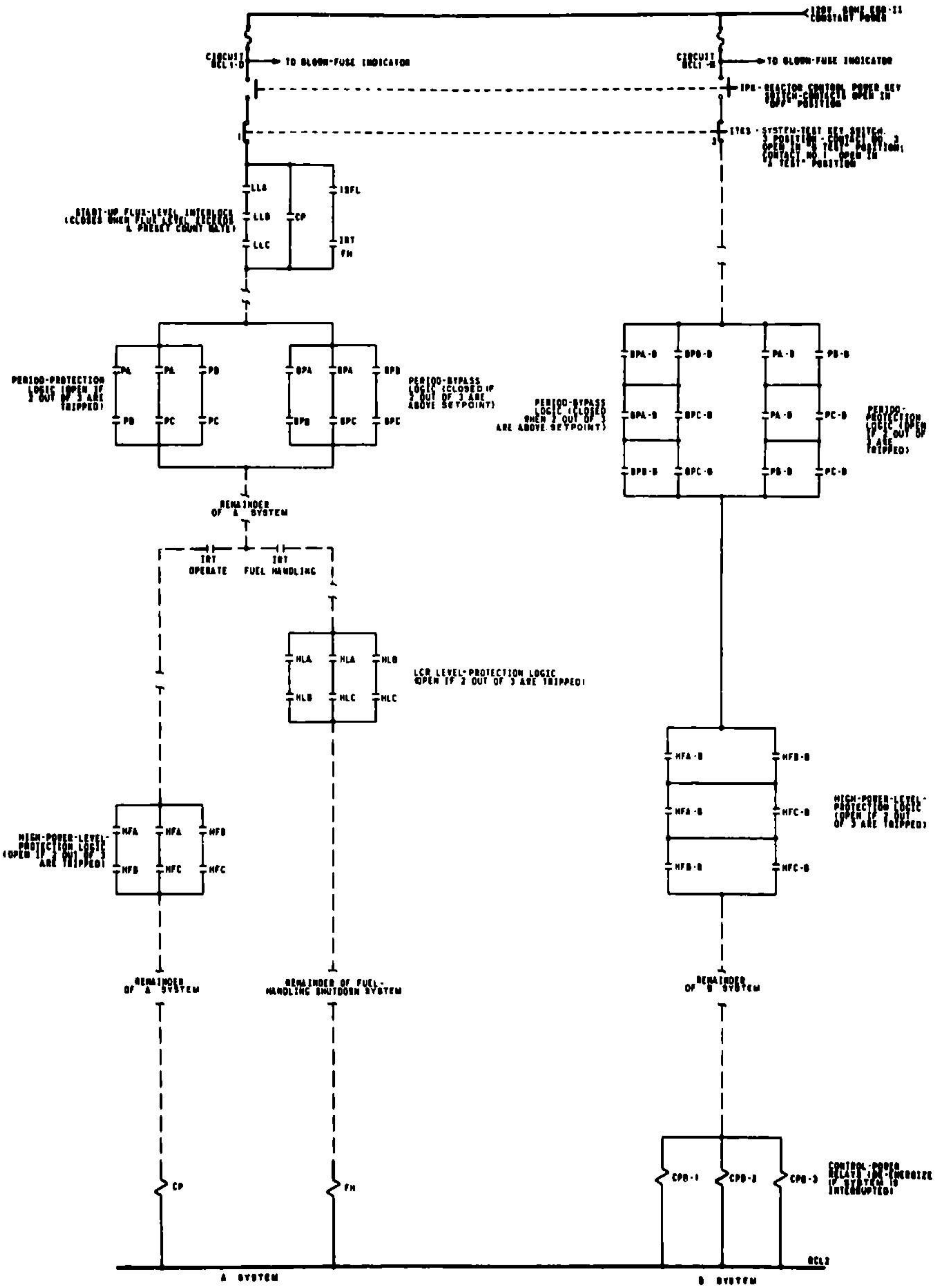


Fig. 14. Reactor Shutdown System after Installation of WAF-753

reactor-operating sections contains the interlocks for startup flux level and period trips. The trip contacts for the startup-flux interlocks are in a one-out-of-three arrangement. This arrangement requires that all three wide-range channels indicate 10 ± 1 cps before the circuit is complete; but once the PPS is energized, the flux interlocks are bypassed. The period trip is bypassed at reactor power above 30 MWt.

The nuclear part of the fuel-handling section of the shutdown circuit contains the LCR high-level trips, and, for channel 7, high-voltage trip, linear level trip, and range 10^{-8} trip. Those for channel 7 require only one trip to start a shutdown. The LCR high-level trips are connected in a two-out-of-three coincidence. The circuit arrangement is such that a shutdown will occur if two or three trips for high count rate occur.

The reactor-operating section of the shutdown system contains the high-flux trips required for power operation. These trips are connected in a two-out-of-three coincidence arrangement so that a shutdown begins if two or three trips for high power level occur. Loss of detector high voltage in a channel will trip both the period and high-flux comparators of the affected channel. This trip automatically places the remaining channels in a one-out-of-two logic configuration for reactor shutdown.

The operating section of the PPS also contains the level trip and high-voltage trip for channel 7, which are normally bypassed with the bypass-key switch for this channel.

3.3.4 Alarm Indications

Two annunciator panels and an array of lighted pushbutton switches in the reactor control room indicate an alarm condition. The W-R nuclear alarms on the shutdown annunciator panel indicate when a reactor trip starts, and those on the nuclear-annunciator panel alarm when a single channel trips. Figure 15 is a diagram of the lighted array of pushbutton switches on the console, which provides both alarm indication and the remote reset station. The trips for high voltage, high LCR, period, LCR-AMS interlock, high flux, and Channel-7 level are reset from this station. The alarms for high flux level, normally set at 104% for full power, are on this array.

3.3.5 Detector

The three guarded fission chambers are in the three operating J thimbles, with Channel A in J-1, Channel B in J-3, and Channel C in J-4. The fission chambers are mounted at an elevation at which the flux is about 1×10^{10} nv at a reactor power of 62.5 MWt.

3.3.6 Preamplifier

Each preamplifier is in a shielded box mounted near the top of a J thimble. Figure 16 is a simplified schematic drawing of the preamplifier.

(SPARE) 1 BK	CH A HIGH FLUX 104% 2 R	CH B HIGH FLUX 104% 3 R	CH C HIGH FLUX 104% 4 R	* CH A HIGH FLUX 110% 5 R	* CH B HIGH FLUX 110% 6 R	* CH C HIGH FLUX 110% 7 R	* CH A HIGH LCR 8 R	* CH B HIGH LCR 9 R	* CH C HIGH LCR 10 R
CH. A-B-C PERIOD BYPASS 11 G	* CH A PERIOD 12 R	* CH B PERIOD 13 R	* CH C PERIOD 14 R	* CH A HIGH VOLTAGE 15 R	* CH B HIGH VOLTAGE 16 R	* CH C HIGH VOLTAGE 17 R	CH A LINEAR TEST 18 R	CH B LINEAR TEST 19 R	CH C LINEAR TEST 20 R
NO. 7 LEVEL BYPASS 21 G	NO. 7 RANGE 1 X 10^{-8} 22 W	* NO. 7 LEVEL 23 BL	NO. 7 HIGH VOLTS 24 R	CH A LOG TEST 25 R	CH B LOG TEST 26 R	CH C LOG TEST 27 R	* CH A LCR-AMS INTERLOCK 28 R	* CH B LCR-AMS INTERLOCK 29 R	* CH C LCR-AMS INTERLOCK 30 R
(SPARE) 31 BK	(SPARE) 32 BK	(SPARE) 33 BK	(SPARE) 34 BK	(SPARE) 35 BK	(SPARE) 36 BK	(SPARE) 37 BK	CH A LOW FLUX 38 R	CH B LOW FLUX 39 R	CH C LOW FLUX 40 R
(SPARE) 41 BK	(SPARE) 42 BK	(SPARE) 43 BK	(SPARE) 44 BK	(SPARE) 45 BK	(SPARE) 46 BK	(SPARE) 47 BK	(SPARE) 48 BK	(SPARE) 49 BK	(SPARE) 50 BK
(SPARE) 51 BK	(SPARE) 52 BK	(SPARE) 53 BK	(SPARE) 54 BK	(SPARE) 55 BK	(SPARE) 56 BK	(SPARE) 57 BK	(SPARE) 58 BK	(SPARE) 59 BK	(SPARE) 60 BK

INDICATES COLOR

BK - BLACK
R - RED
G - GREEN
W - WHITE
BL - BLUE

* PUSHBUTTON PROVIDES RESET FUNCTION

Fig. 15. Wide-range Control and Alarm Panel

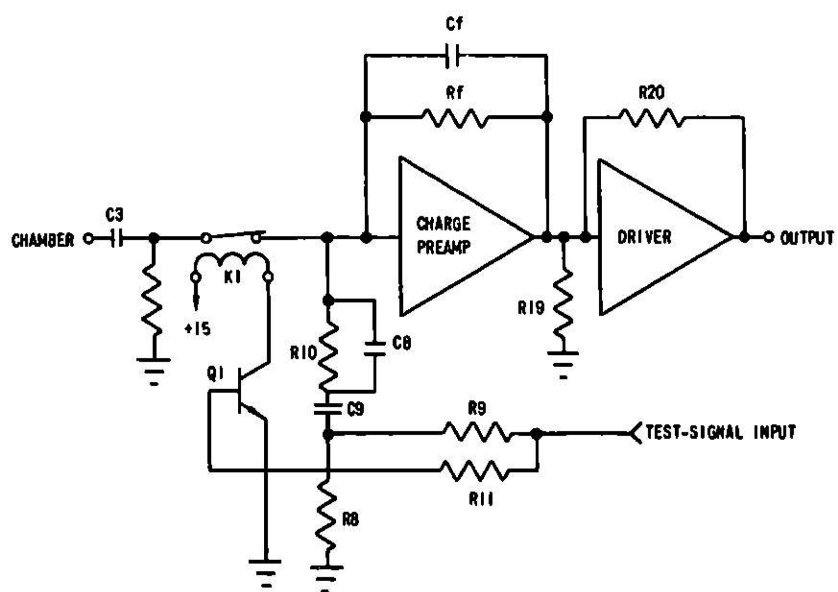


Fig. 16. Schematic of PA-5 Preamplifier

3.3.7 Wide-range Chassis

Each W-R monitor is in a separate cabinet in the cable-routing room (CRR). Figure 17 is a diagram of the CRR layout, showing the locations of the W-R monitor cabinets. Each monitor consists of a log chassis (see Fig. 18) and a linear drawer (see Fig. 19).

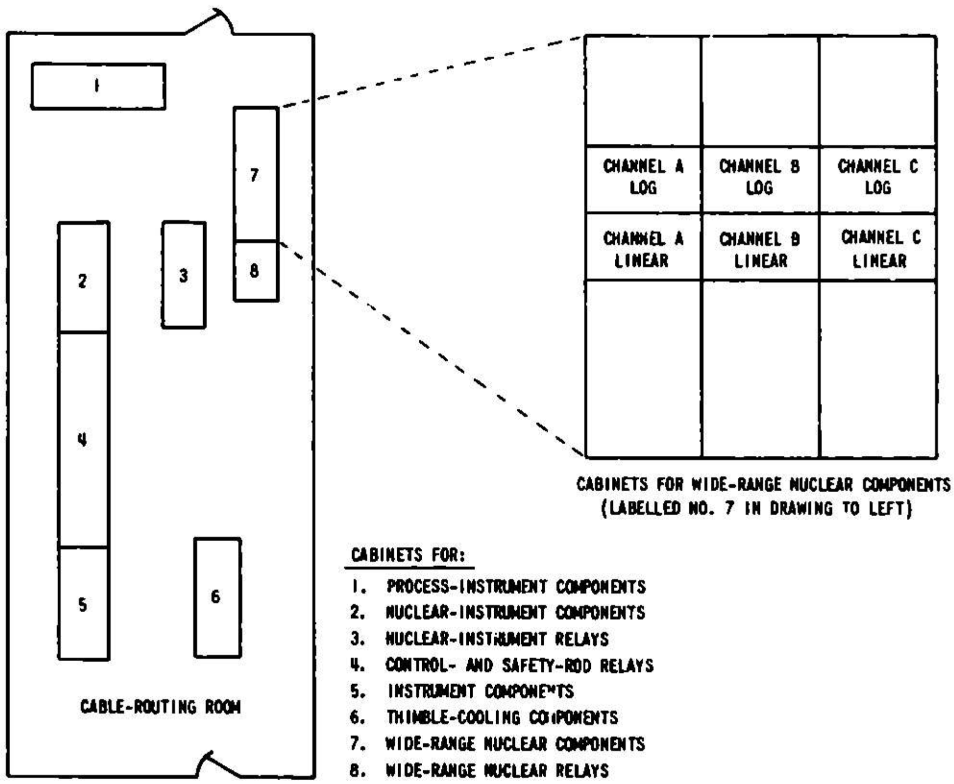


Fig. 17. Locations of Wide-range-channel Equipment in Cable-routing Room

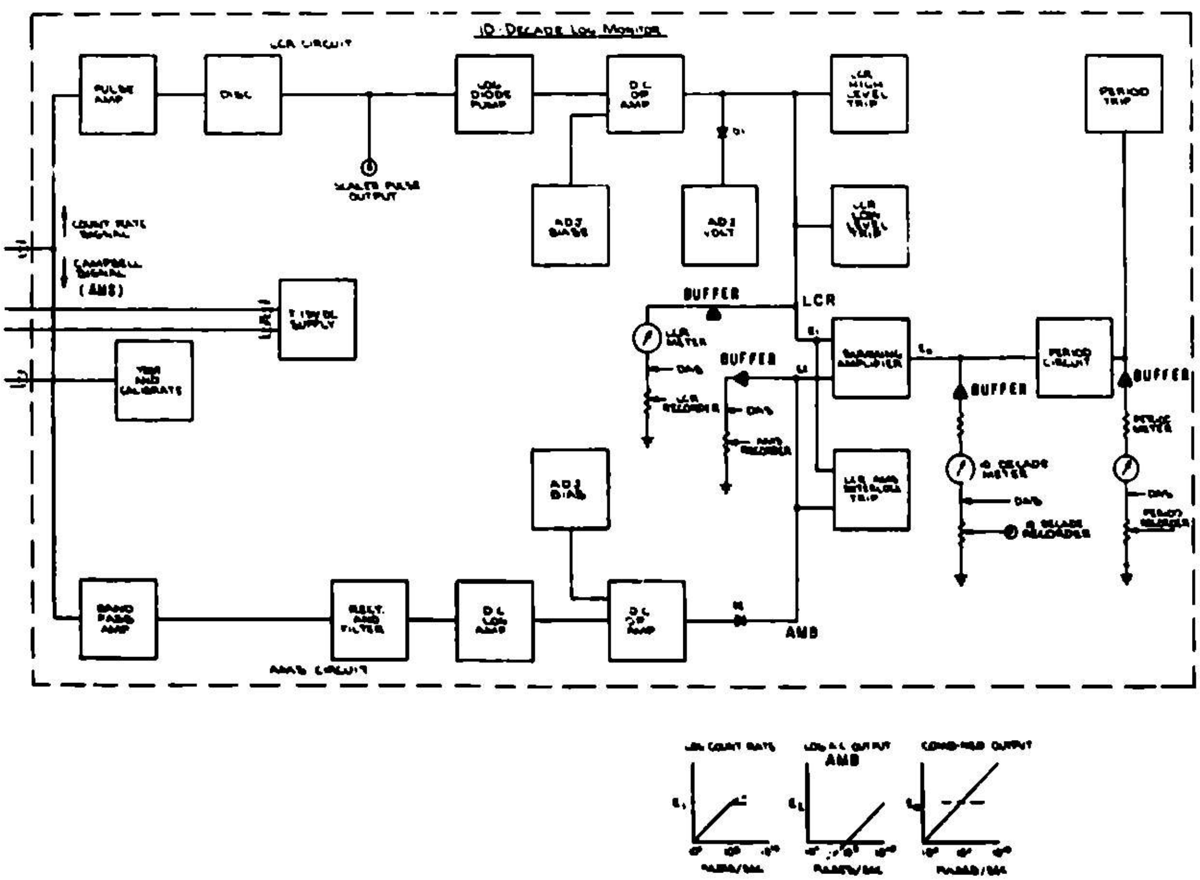


Fig. 18. Wide-range-channel LCR Circuit. ANL Neg. No. 103-A9562.

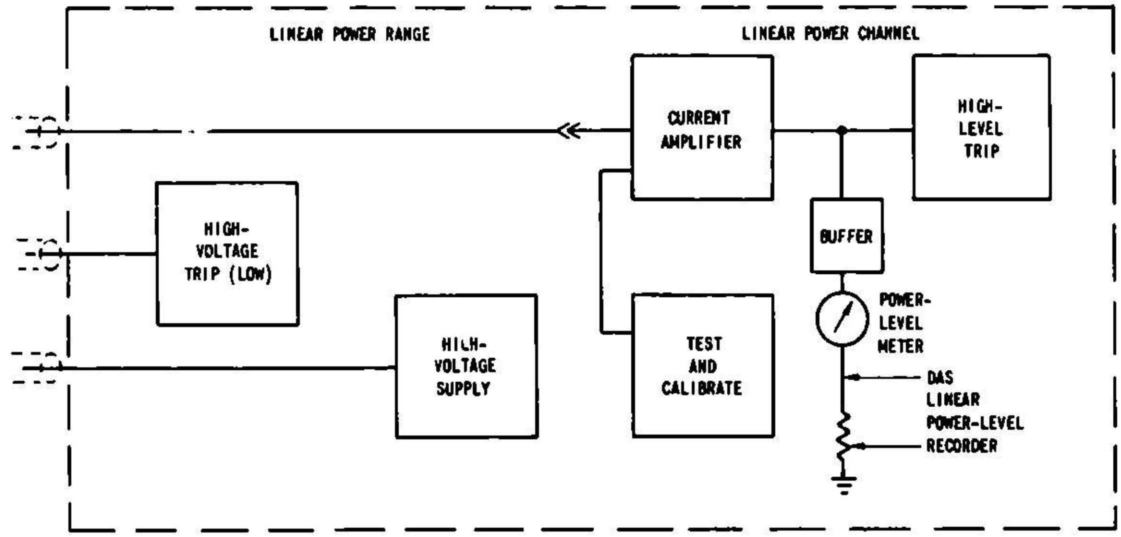


Fig. 19. Wide-range-channel Linear Circuit

The W-R monitors located in the three cabinets are separated to meet the independence requirements of RDT Standard C16-1T.[4]

3.3.8 Shutdown Relays

The W-R nuclear shutdown relays are in a cabinet (see Fig. 17). The relay cabinet is divided into three compartments by fire barriers to satisfy the separation requirements of RDT C16-1T.[4] A second set of relays has been installed as part of shutdown system B (see Sec. 2). This upgrading of the circuits of the shutdown system to meet requirements of RDT C16-1T is not considered part of this modification of the W-R channel.

3.3.9 Normal Performance Characteristics

The W-R channels are in operation during all phases of reactor operation and provide indication, alarm, and trip functions. Figure 20 shows the operating ranges of the channels. The LCR, AMS, and linear-power-range circuits constitute the measuring circuits; each provides a readout. The 10-decade signal is the electronic sum of the LCR and AMS signals. The ranges are shown for the fission counter/chamber positioned in a flux of 10^{10} nv at 62.5 MWt. The 10-decade and AMS readouts then indicate 100% power. The circuit for linear power range gives an output of 1.55 mA at 10^{10} nv; the circuit is calibrated to indicate 62.5% of full range on its readout device.

The LCR, AMS, and 10-decade signals are logarithmic. The LCR and AMS signals overlap by at least $1\frac{1}{2}$ decades. The 10-decade signal is composed of the AMS and LCR signals, which are electronically summed to provide a smooth response in the crossover region. The power-level circuit provides a linear output with four decades of manual range changing.

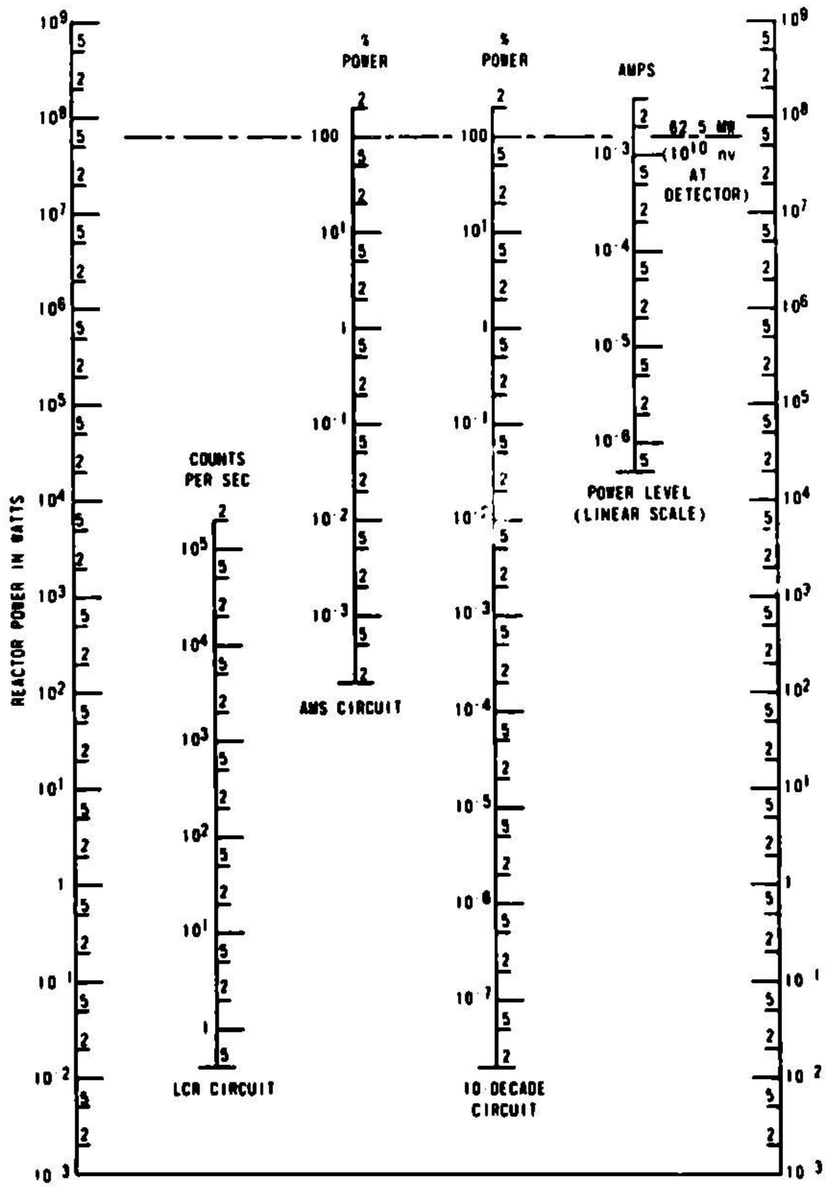


Fig. 20. Ranges of Wide-range Nuclear Channels

3.3.10 Requirements for Accuracy and Response Time

The required values for accuracy and response time of the W-R system, as they relate to limiting-safety-system settings and limiting conditions for operations, are as follows:

a. Period-trip accuracy is within 8 s at a normal trip setting of 25 s.

b. Time response for period trip (<40% full power) is less than 0.37 s for a simulated step increase from 12 to 90 kW with setpoint at 25-s period.

c. Accuracy of high-flux-level trip is within 5% at a normal trip setting of 110% of full power.

d. Time response for high-flux-level trip is less than 0.1 s with a simulated step increase in power from 100 to 125% with setpoint at 115%.

For unrestricted fuel handling, the values are:

a. Time response for period trip is less than 0.22 s for a simulated step increase from 100 to 10^4 cps with the setpoint at 25-s period.

b. Time response for high count-rate level is less than 0.1 s for a simulated step increase from 100 to 10^4 cps with the setpoint at 1500 cps.

3.3.11 <u>Calibration, Test, and Trip Setup</u>

The calibration, test, and trip setup of each channel were performed when the systems were installed and will be performed periodically during maintenance and reactor operation. Completely independent test and calibration circuits are in each log and linear-power drawer to allow channel test and calibration checks without removing the instruments. Because the redundant channels operate on two-out-of-three logic into the protection system, any one channel can be temporarily tested or calibrated without causing a reactor trip. Annunciators make the reactor operator aware that a channel is being calibrated. When any log drawer or linear drawer is being calibrated, the period relay of the log channel or the high-level relay of the linear channel is also automatically tripped. The tripped relays and annunciators clear when the CALIBRATE switches are returned to the OPERATE position.

The log drawer has a period test signal that adds a signal to the chamber signal. The period relay is not automatically tripped by rotating this control until the setpoint is exceeded. Local indication that this control is not in the OPERATE position is provided by a lighted NONOPERATE light on the log chassis. The linear level trip can be tested by rotating the test control. This rotation adds a signal to the chamber signal and will not cause a linear level trip until the trip setpoint is exceeded. That this control is not in the OPERATE position is indicated locally by a lighted NONOPERATE light on the linear drawer.

The setpoints in the log and linear chassis are verified before reactor startup and before unrestricted fuel handling. When the linear-power-level trip is being verified, the period-bypass-relay of that channel is energized. Because the period-bypass logic is two out of three, protection is still provided should the control be inadvertently left in the up-scale position. During test and verification of trip levels with the period-bypass relay energized, a single fault in the level-trip circuitry could result in a bypass of the period-trip circuitry. Such an occurrence would be annunciated as a period-trip bypass and would not constitute an undetected failure. If the reactor power is below 25 MWt, the operator must manually trip the reactor upon the receipt of a valid indication of period-trip bypass.

In general, operation of the ^{235}U detectors in the W-R channels can be verified before reactor startup because the LCR circuit is sensitive enough to measure neutrons with the reactor shut down.

The circuit for low-count-rate trip is tested before each reactor run and before unrestricted fuel handling by a built-in test circuit in each chassis. In addition, whenever work is done on the detectors that could affect their sensitivity, the reactor source is removed from the core and background counts are taken on each channel to verify that the events detected are neutron pulses and not noise or gamma background. Proper response to changes in neutron levels can also be verified by moving the safety rods in the core and changing the neutron multiplication factor.

During calibration or test, function switches on the panels of the chassis switch the channel from the operate condition to various calibrate and test conditions. During a level calibration or trip test, the rate circuit is inhabited and remains until the test is completed and the function switch has returned to the OPERATE position.

3.3.12 Time-response Characteristics of W-R Channels

3.3.12.1 Response of Linear Power Circuit. The linear power circuit consists of a fission chamber, linear-current amplifier, and trip circuit Detailed information on fission-chamber response is not available; however, ion-collection time in the chamber is less than 1 μs; therefore the chamber does not conbribute significantly to the overall circuit response.

The linear-current amplifier has a time constant (0-63%) of 500 μs. The trip comparator has a time constant of 500 μs and an 8-ms delay due to relay release time.

For reactor periods longer than about 10 ms, the instrument closely follows reactor power and produces a trip 8 ms after the setpoint has been reached. For reactor periods shorter than 10 ms, the time constants of the linear amplifier and trip comparator delay the trip.

3.3.12.2 Response of Log-N Period Circuit. Response of the wide-range-channel period circuit was calculated by using the Continuous System Modeling Program (CSMP).[7] Results for startup cases are listed in Table II, and those for at-power cases in Table III.

Figure 21 shows the model used to determine instrument response, including initial conditions and time constants. Time to reach the setpoint is indicated by a change in output of the state of the comparator. For relays and control-clutch release times, total time to trip is calculated by adding 58 ms to the time to reach the setpoint.

TABLE II. Parametric Analysis for Reactivity Insertions at Startup[a]

Linear Ramp Insertion (Unbounded), $/s	Time to Period Trip,[b] s		Time to Power Trip[b] 115%, s	Time to Temperature Trip[b] 115%, Row 2, s	Peak Cladding Temperature Reached,[c] Mark-II Fuel, °F[d]			Time to Severity Levels, Mark-II Fuel,[c] s		Time to 1834°F, Fuel Melting, s	Comments
	17-s Period	25-s Period			Period Trip; 17-s Period	Power Trip, 115%	Temperature Trip, 115%	1319°F	1500°F		
0.001	292.1	235.1	444.1	451.8	700.5	1163	1196	495.7	603.1	560.5	
0.01	28.5	21.0	78.1	80.6	700.0	1159	1267	82.0	88.0	85.6	
0.1	4.22	3.51	10.7	12.6	700.0	1085	1603	11.5	12.2	11.9	Base cases: Inlet coolant temperature = 700°F.
1.0	1.02	0.92	1.081	-	702.3	831		1.395	1.501	1.446	
10.0	0.242	0.221	0.167	-	>1500	1387		0.160	0.178	0.139	
0.001	292.1	235.1	444.1	442.0	715.5	1175	1178	489.2	593.2	555.9	
0.01	28.5	21.0	78.1	79.8	715.0	1170	1247	81.6	87.4	85.4	
0.1	4.22	3.51	10.7	12.5	715.0	1100	1596	11.4	12.1	11.8	Inlet coolant temperature = 715°F.
1.0	1.02	0.92	1.081	-	717.3	846		1.386	1.493	1.444	
10.0	0.242	0.221	0.167	-	>1500	1400		0.159	0.177	0.138	
0.001	292.1	235.1	443.2	588.3				582.0	705.0	622.0	
0.01	28.5	21.0	78.1	89.1				86.8	54.5	91.2	
0.1	4.22	3.51	10.7	13.7				11.96	12.8	12.3	Reactor startup with inlet coolant temperature at 580°F.
1.0	1.02	0.92	1.081	-				1.48	1.59	1.50	
10.0	0.242	0.221	0.167	-				0.174	0.192	0.145	

[a] Initial conditions: just critical; neutron power = 62.5 MWt.
[b] Time when trip rods begin to move; includes instrument response and delays.
[c] Without uncertainties.
[d] Conversion factor: °C = (°F − 32)/1.8.

TABLE III. Parametric Analysis for Reactivity Insertions at Power

Linear Ramp Insertion (Unbounded), $/s	Time to Period Trip,[a] s		Time to Power trip[a] 115%, s	Time to Temperature Trip[a] 115%, Row 2, s	Peak Cladding Temperature Reached,[b] Mark-II Fuel, °F			Time to Severity Levels, Mark-II Fuel,[b] s		Time to 1834°F, Fuel Melting, s	Comments
	17-s Period	25-s Period			Period Trip, 17-s Period, °F[c]	Power Trip, 115%, °F	Temperature Trip, 115%, °F	1319°F	1500°F		
0.001	No trip[d]	No trip	64.1	71.8		1169	1177	181.3	312.6	324.4	
0.01	No trip	No trip	9.97	12.9		1166	1189	26.0	40.5	41.8	
0.02	No trip	No trip	5.66	8.04		1164	1197	14.8	22.6	23.2	
0.04	16.31	8.85	3.13	5.39	>1500	1161	1218	8.43	12.6	12.96	
0.06	8.80	6.25	2.20	4.22	1487	1158	1231	6.07	9.02	9.24	Base cases: Inlet coolant temperature = 700°F.
0.1	5.67	4.37	1.40	3.30	1477	1152	1264	4.01	5.89	6.03	
0.2	3.48	2.76	0.738	2.48	>1500	1141	1347	2.31	3.31	3.38	
0.4	2.26	1.83	0.395	1.94	>1500	1131	>1500	1.35	1.88	1.91	
1.0	>1.16	1.08	0.190		>1500	1122		0.698	0.916	0.918	
10.0	>0.22	>0.22	0.071		>1500	1137		0.114	0.131	0.117	
0.001	No trip	No trip	64.2	38.0		1184	1156	169.9	301.7	317.9	
0.01	No trip	No trip	9.97	7.57		1181	1164	24.6	39.4	41.1	
0.1	5.67	4.37	1.40	2.41	1493	1168	1221	3.82	5.74	5.95	Inlet coolant temperature = 715°F.
1.0	1.16	1.08	0.190	1.28	>1500	1138	>1500	0.675	0.899	0.909	
10.0	>0.22	>0.22	0.071		>1500	1152		0.112	0.129	0.116	
4.65 (Finite insertion, 0.33$)	No trip	No trip	0.084	1.60	e	1172	1287	3.25	e	e	Finite reactivity insertion of 0.33$, inlet coolant temperature = 700°F.

[a] Time when trip rods begin to move; includes instrument response and delays.
[b] Without uncertainties.
[c] Conversion factor: °C = (°F − 32)/1.8.
[d] Minimum reactor period reached is greater than trip level.
[e] Maximum temperatures reached after 25 s are: 1455°F (cladding temperature), and 1749°F (fuel temperature).

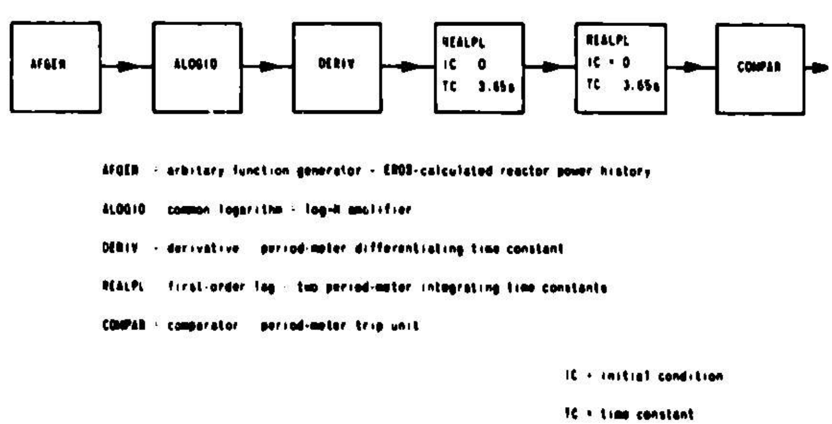


Fig. 21. Model of Period Circuit of Wide-range Channels for Use in CSMP

At the start of the transient, the log-N circuit has a time constant of 40 ms, which reduces to a constant 15 ms after about three decades of power increase. Compared with the two time constants for the period circuit, this time constant is small and can be ignored.

3.3.13 Precision and Accuracy of Channels

3.3.13.1 Precision of Linear Power Circuit. Characteristics of the linear-power-circuit components, except for the fission chamber, appear in Ref. 8. From the published saturation characteristics, the chamber is expected to contribute less than ±0.1% to the inaccuracy of the circuit.

Precision of the linear-current circuit is given in terms of linearity and long- and short-term drift. Accuracy of the trip comparator is expressed in terms of resolution and reproducibility.

Overall circuit precision can be expressed as

$$\text{Precision} = \pm \sqrt{A_d^2 + L_a^2 + D_{a1}^2 + D_{a2}^2 + R_t^2 + D_t^2},$$

where (in percent of full scale)

A_d = Detector linearity = ±0.1,

L_a = Amplifier linearity = ±0.5,

D_{a1} = Amplifier drift = ±0.5%,

D_{a2} = Amplifier Long-term drift = ±0.2,

R_t = Trip resolution = ±0.5%,

D_t = Trip drift (reproducibility) = ±0.2,

and

$$\text{Precision} = \pm 0.92\%.$$

Full scale for the linear power circuit is about 125 MWt; therefore, precision expressed in terms of reactor power is ±1.15 MWt, or ±1.8% of full reactor power.

The nominal setpoint for the linear power circuits is 110% of power. The worst-case setpoint is therefore 111.8% of power as determined from thermal calibration.

3.3.13.2 Accuracy of Log-N and Period Circuits. Accuracy of the 10-decade log-N circuit is specified as a linearity of ±1.5% of linear-equivalent full scale.

Accuracy of the period circuit is ±1% of linear-equivalent full scale. The period-circuit full scale is 10 V, corresponding to a +10-s period. Accuracy of the trip comparator is ±0.7% of full-scale 10 V. Combining the period-circuit accuracy of ±0.1 V and the comparator accuracy of ±0.07 V and applying them to the nominal 25-s setpoint yield

$$\text{Setpoint} = 4 \pm 0.17 \text{ V} = 25 \pm 1 \text{ s}.$$

Nominal conversion gain of the log-N circuit is 1 V/decade. If we assume that a worst-case nonlinearity of the circuit can occur within one decade, the conversion gain will be reduced by 30% within that decade. With a worst-case setpoint of 24 s and a 70% conversion gain, the actual period required to produce a trip will be 17 s.

3.3.14 Reliability of Channels

The W-R channels procured from Gulf are of high quality; the components either meet military specifications or are comparable. The quality-assurance program at Gulf for this procurement was reviewed and approved. The program contained sufficient testing, checks, and control points to ensure adequate quality of both vendor-supplied components and fabricated assemblies.

The channels are the same as Gulf's standard catalog equipment, except for some engineering changes required to meet the requirements of RDT Standard C15-2T[9] and the EBR-II ordering data. Gulf has extensively tested the standard W-R channels to eliminate improper design. Complete tests of temperature, humidity, line voltage, and line frequency were made to ensure that specifications have been met. Tests with transient neutron pulses were also made to show that channels do not fail or have fold-over, because of rapidly increasing power rate up to 4×10^{13} nv at the detector with periods as short as 4 ms.

The W-R monitors manufactured by Gulf have accumulated 400,000 operating hours. Included is time accumulated on power and Triga type of reactors. All the linear-power-range monitors fabricated by Gulf have accumulated 583,000 operating hours. From these figures, Gulf calculated the mean time between failure (MTBF) in either the safe or unsafe direction as 190,000 h for the wide-range monitor and 250,000 h for the linear-power-range monitor. The values are stated for a confidence factor of 90%. The MTBF values given above are low if only unsafe failures are considered. Safe failures provide a trip in the coincidence logic and annunciation, and therefore do not reduce system safety.

With these MTBF values, the probability of the system (three channels) not providing protection during a reactor run is developed below. The equation for failure probability of two-out-of-three systems has been developed.[5]

The following system-failure probabilities are based on all failures, both safe (detectable) and unsafe (undetectable). Therefore, the probabilities of unsafe failures are expected to be considerably less than calculated.

3.3.14.1 Gulf Linear Channel. Gulf Electronic Systems quote an estimated MTBF of 250,000 h with a 90% confidence. This number is based on 583,000 actual hours of operating experience. The number of failures per hour is

$$\lambda = \frac{1}{\text{MTBF}}$$

$$= \frac{1}{250{,}000 \text{ h}}$$

$$= 4 \times 10^{-6}/\text{h}.$$

If a time interval (T) equal to a one nominal reactor run of 600 h is used, the failure probability for the two-out-of-three system is

$$P_f = 3(\lambda T)^2 - 2(\lambda T)^3$$

$$= 3(4 \times 10^{-6} \times 600)^2 - 2(4 \times 10^{-6} \times 600)^3$$

$$= 5.8 \times 10^{-6}.$$

For a reactor run of 2000 h, which is longer than any expected, the failure probability then is

$$P_f = 3(4 \times 10^{-6} \times 2000) - 2(4 \times 10^{-6} \times 2000)^3$$

$$= 6.3 \times 10^{-5}.$$

3.3.14.2 Gulf Wide-range Channel. Gulf Electronic Systems quote an estimated MTBF of 190,000 h with a 90% confidence. This number is based on 400,000 actual hours of operating experience. The number of failures per hour is

$$\lambda = \frac{1}{\text{MTBF}}$$

$$= \frac{1}{190{,}000 \text{ h}}$$

$$= 5.3 \times 10^{-6}/\text{h}.$$

With T = 600 h,

$$P_f = 3(5.3 \times 10^{-6} \times 600)^2 - 2(5.3 \times 10^{-6} \times 600)^3$$

$$= 3 \times 10^{-5}.$$

For a reactor run of 2000 h, which is longer than any expected, the failure probability then is

$$P_f = 3(5.3 \times 10^{-6} \times 2000)^2 - 2(5.3 \times 10^{-6} \times 2000)^3$$

$$= 3.3 \times 10^{-4}.$$

3.3.15 Design Features to Mitigate Effects of Casualty Events

3.3.15.1 Loss of Thimble Cooling. Loss of thimble cooling would result in the detector-thimble temperatures increasing above the maximum allowable value for the detector and detector cables. The thimble-cooling system is designed to alarm on several abnormal conditions, including high temperature. The operator action required in the event of abnormal thimble cooling allows time to remove the detectors before they are damaged.

3.3.15.2 Loss of Nuclear Constant Power. Loss of nuclear constant power will result in loss of indication from the nuclear channels. Protection will not be sacrificed because the trip relays in the nuclear chassis will deenergize on loss of power and thus trip the reactor.

3.3.15.3 Radiation Damage to Detector Cable. The detector cables are subjected to neutron and gamma fluences that eventually damage the cable insulation. Dielectric strength is reduced, resulting in extreme noise, leakage currents, and electrical breakdown. Noise at startup causes increased count rate and an unstable period indication. High noise levels may cause LCR level trips and period trips. At power, the leakage current causes a higher than normal indication on the linear level recorders. Extreme leakage will result in a level trip.

3.3.15.4 Damage from Fire. Fire in the reactor building or power plant could damage the W-R cables and result in an inoperative channel or system. The cables are separated as far as practicable to reduce the probability of system failure. The cables for Channels A-C are in separate conduits or raceways, except where two channels share one reactor-building penetration. Only two coaxial-cable-type penetrations exist; installing another one was considered impracticable when only low-level signals are carried by cables in these penetrations, which are separated by fire barriers. The W-R chassis are so located that each channel is in its own cabinet. The shutdown PPS relays are in a cabinet with fire barriers and wired with fire-retardant wire.

3.3.15.5 Abnormal Temperature in Cable-routing Room. Extreme abnormal temperatures in the cable-routing room (CRR) could cause the W-R equipment to operate outside its design limits. Temperature in the CRR is thermostatically controlled, and the room temperature is observed once each shift. Temperature detectors are also mounted in the W-R channel cabinets to sound an alarm in the control room if any cabinet temperature exceeds design limits.

3.3.15.6 Abnormal Temperature and Humidity at Preamplifiers. Abnormal temperatures at the preamplifier locations in the reactor building can also cause the W-R equipment to operate outside design limits. Controls, alarms, and monitoring similar to those in the CRR are provided.

Abnormal humidity at the preamplifier locations can range from 10 to 90%. This range does not exceed the design limits of the W-R equipment.

3.4 Justification for Modification

This modification is to upgrade EBR-II reactivity protection by:

a. Providing neutron-flux signals less sensitive to gamma-flux levels.

b. Providing greater overlap between startup and intermediate-range flux signals, while retaining complete overlap of intermediate and power-range flux signals.

c. Reducing the number of trip contacts in the PPS.

d. Reducing the susceptibility for spurious trips during fuel handling.

e. Providing a reactivity protection subsystem that meets the requirements of RDT C16-1T[4] insofar as practicable.

3.5 Applicable Standards

This modification meets the requirements of the following standards except for specific variances, which are discussed below:

Number	Title
RDT C16-1T	Supplementary Criteria and Requirements for RDT Reactor Plant Protection Systems[4]
RDT C15-2T	Wide-range (10-decade) Neutron Flux Monitoring Channel[9]

The cables from the preamplifiers for Channels A and B pass through the walls of the reactor building through connectors in a common penetration, because only two coaxial-cable-type penetrations are installed. Therefore, a single event could conceivably affect these two channels. However, a metal barrier is between the connectors for Channels A and B, and the cables are in separate conduits where they penetrate the barrier. Thus the intent of the physical-separation requirements of RDT C16-1T is being met for fire protection, and a fire is considered the only credible event that might damage the cables in the common penetration. This minor variance with RDT C16-1T could not be avoided without installing a new penetration; such an installation is considered economically impracticable.

3.6 Safety Analysis

3.6.1 Safety Limits and Performance Requirements

3.6.1.1 Fuel-element Integrity. Safety limits have been developed to ensure fuel-element integrity in the event of postulated design-basis transients, including reactivity insertion. The limits, intended to ensure that loss of cladding is of low probability and does not constitute a safety hazard, are: Power-to-flow ratio upon identified reactor faults shall be limited so that the fuel-cladding interface temperature of Mark-II driver fuel does not exceed 1319°F (715°C) for more than 60 s and in no case exceeds 1500°F (810°C). These limits are intended to preclude cladding loss greater than 5% due to eutectic formation between fuel and cladding.

3.6.1.2 Design-basis Reactivity Transients. Limiting anticipated, unlikely, and extremely unlikely faults have been defined for EBR-II and provide the design-basis transients for analysis of reactor operations and irradiations of experimental subassemblies. These faults are listed in Table IV. The analysis of the time response of the modified PPS to these postulated transients[5,6] is discussed in Sec. 3.6.2. Several reactivity-insertion events may be hypothesized in which malfunctions follow the initial fault (e.g., stuck control rod); however, none have any significant effect on the safety margins.[10]

TABLE IV. Design-basis Reactivity Transients for Analysis of EBR-II PPS Requirements

Category	System Fault
Anticipated	Single control-rod insertion at startup. Single control-rod insertion at power.
Unlikely	Two control rods inserted at startup. Two control rods inserted at power. Safety rods inserted at startup.
Extremely unlikely	More than two high-worth control rods inserted at startup or at power.
	Dropping of an instrumented subassembly at power.
	Reactivity insertion with failure of PPS to act.
	Desing-basis accident assuming melting of the core with subsequent rapid compaction of core under gravitational effects.

3.6.1.3 Performance Requirements. Limiting setpoint requirements for reactor power level and period trips have been established:

Period trip (from Critical to 25 MWt)	17 s
Power-level trip	115% of full power

Those requirements are established for off-normal power conditions that may be hypothesized to occur slowly (for example, from operator error in adjustment of a control rod). They ensure that the lower temperature bound for formation of fuel-cladding eutectic will not be exceeded and the 60-s time limit for temperatures above that for eutectic formation will therefore not be a factor.

Time-response requirements of PPS instrumentation are determined from the hypothesized faults resulting in rapid temperature transients. For these faults, two safety-limit bounds are applied:

a. Eutectic-formation temperatures must not be exceeded for more than 60 s.

b. Peak fuel-cladding interface temperatures must not exceed 1500°F (815°C).

3.6.2 Performance of Modified System

3.6.2.1 Response to Reactivity Insertions. Table V summarizes the response of the EBR-II PPS to the anticipated and unlikely design-basis transients (see Table IV) with the W-R channels installed. Listed are the times to trip, time required to reach temperature limits without trip, and the protective margins in seconds and °F. Figures 22-24 show parametric relationships between reactivity insertion rate, time, power, period, and fuel temperatures for startup conditions, full power, and fuel handling, respectively. The table and figures show that the power level, temperature, or period trips effectively terminate the postulated reactivity transients before temperature limits at the fuel-cladding interface are reached.

TABLE V. Response of EBR-II PPS to Anticipated and Unlikely Reactivity Transients

Reactivity Insertion	Initial Reactor Power	Time to Power Trip,[a] s	Time to Temp Trip,[a] s	Time to 1290°F[b] Mark-IA, 1319°F Mark-II, s		Time to 1500°F, s		Time to 1641°F, s		Time to 1834°F, Fuel Melting, s	Peak Cladding Temp: Power Trip, °F		Peak Cladding Temp: Temp Trip, °F		Protective Margin: Time, s		Protective Margin: Temp, °F	
				w/o[c]	w[d]	w/o	w	w/o	w		w/o	w	w/o	w	Power Trip	Trip[e]	Power Trip	Temp Trip[e]
Two Control Rods	SU[f]	51.34	53.47															
Mark-IA				53.6	52.6	57.3	55.3	60.5	57.6	57.2	1113	1178	1295	1388	1.26	1.83	322	117
Mark-II				53.5	52.4	56.3	54.5	59.0	56.5	55.2	1144	1234	1340	1440	1.06	1.03	286	60
Safety Rods	SU	207.49	210.24															
Mark-IA				219.1	212.6	239.4	228.3	255.3	240.8	238.8	1133	1201	1177	1252	5.11	18.06	299	248
Mark-II				217.8	211.8	233.8	223.8	247.9	234.0	227.7	1164	1237	1211	1291	4.31	13.56	263	209
One Control Rod	SU	85.58	88.09															
Mark-IA				90.1	87.8	98.6	93.7	106.4	99.4	98.3	1120	1186	1241	1326	2.22	5.61	104	174
Mark-II				89.6	87.4	96.0	91.9	102.7	96.6	93.5	1151	1222	1281	1372	1.82	3.81	97	128
Two Control Rods	100%	7.99	10.73															
Mark-IA				19.8	14.6	30.5	25.3	36.5	31.1	33.1	1136	1204	1163	1236	6.61	14.57	296	264
Mark-II				19.2	13.9	28.2	23.1	34.1	18.7	28.9	1166	1239	1195	1273	5.91	12.37	261	227
One Control Rod	100%	13.54	16.63															
Mark-IA				33.0	24.1	51.4	42.4	62.5	52.5	56.1	1137	1206	1156	1228	10.56	25.77	84	272
Mark-II				32.0	23.1	47.3	38.5	57.9	48.5	48.5	1167	1240	1186	1262	9.56	21.87	79	238

[a]Time when trip rods begin to move.
[b]Conversion factor °C = (°F - 32)/1.8, for protective margins, °C = °F/1.8.
[c]w/o = Without uncertainties.
[d]w = With uncertainties.
[e]Protective margins for subassembly-temperature trips are based on the severity level for unlikely faults (1500°F).
[f]Startup.

Reactivity insertions due to potential refueling accidents are discussed under Cases 1, 2, 5, and 6 of Appendix F of Ref. 2. These data and Fig. 24 show that a safety-rod trip initiated from a trip of either the short-period or high-counting-rate circuits of the W-R channels would provide protection under the conditions of Cases 1, 2, and 6, but not under the conditions of Case 5. The reason for this is that the safety rods could not remove reactivity fast enough to prevent fuel melting, even if the channel trips occurred when the central subassembly was dropped. This hypothetical unprotected accident, as discussed in Ref. 2, requires a very improbable set of conditions and is an "extremely unlikely accident" under the RDT C16-1T classification system. Therefore, the considerations for fuel-handling accidents discussed in Ref. 2 are applicable to the system as modified by installation of the W-R channels.

For a reactivity-insertion fault, with reduced shutdown reactivity available, i.e., one highest-worth control rod stuck, sufficient shutdown

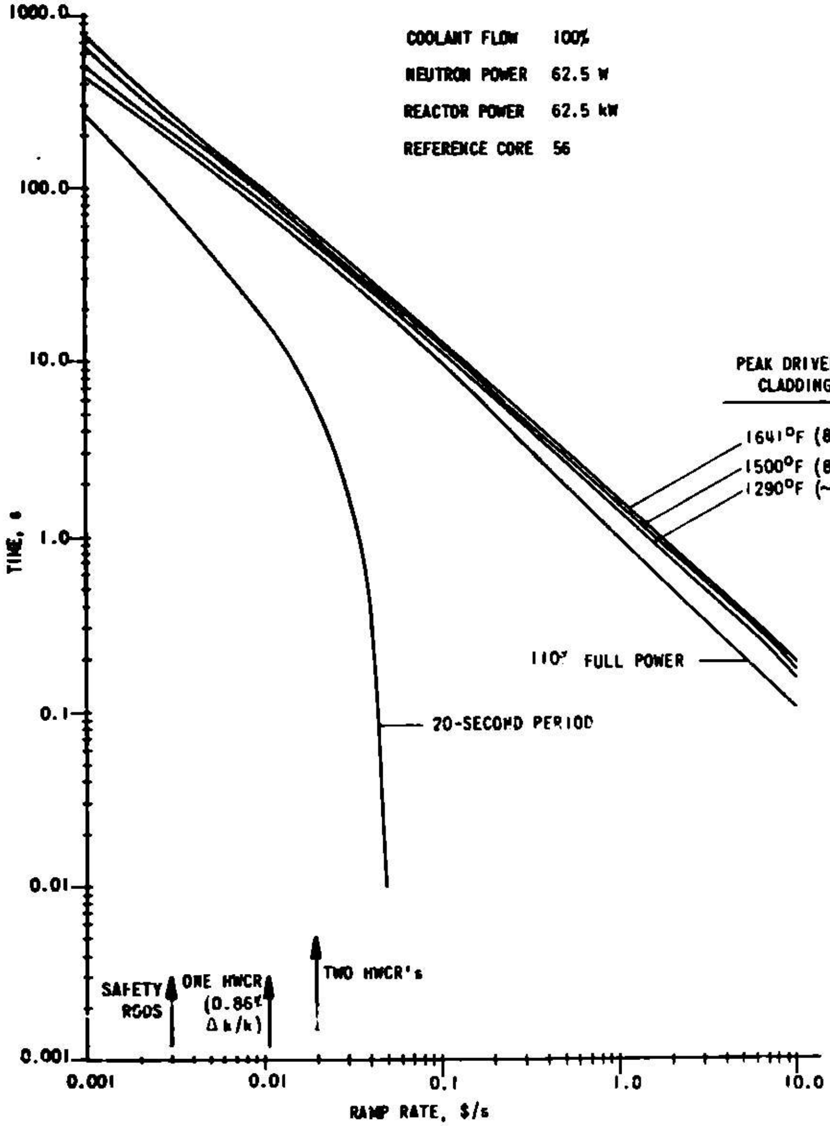


Fig. 22. Time vs Reactivity Ramp Insertion Rate; Initial Conditions: Startup

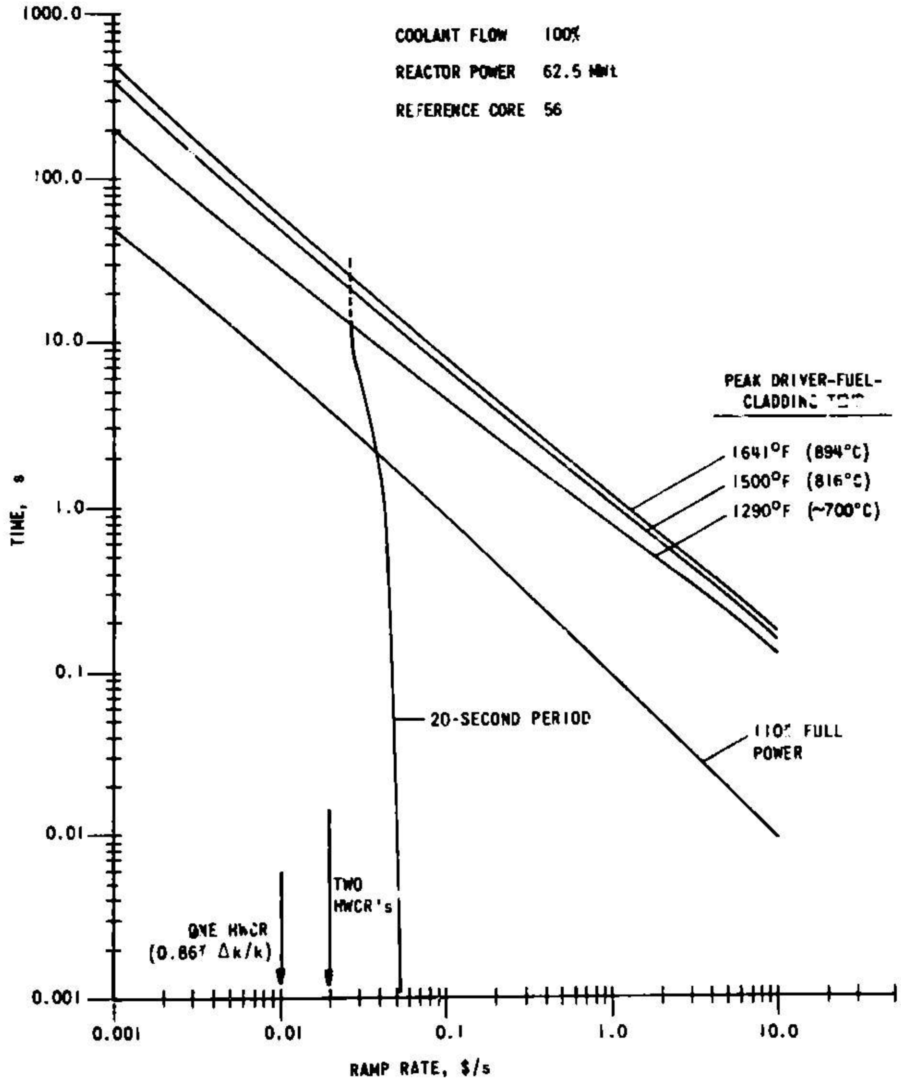


Fig. 23. Time vs Reactivity Ramp Insertion Rate; Initial Conditions: Power Operation

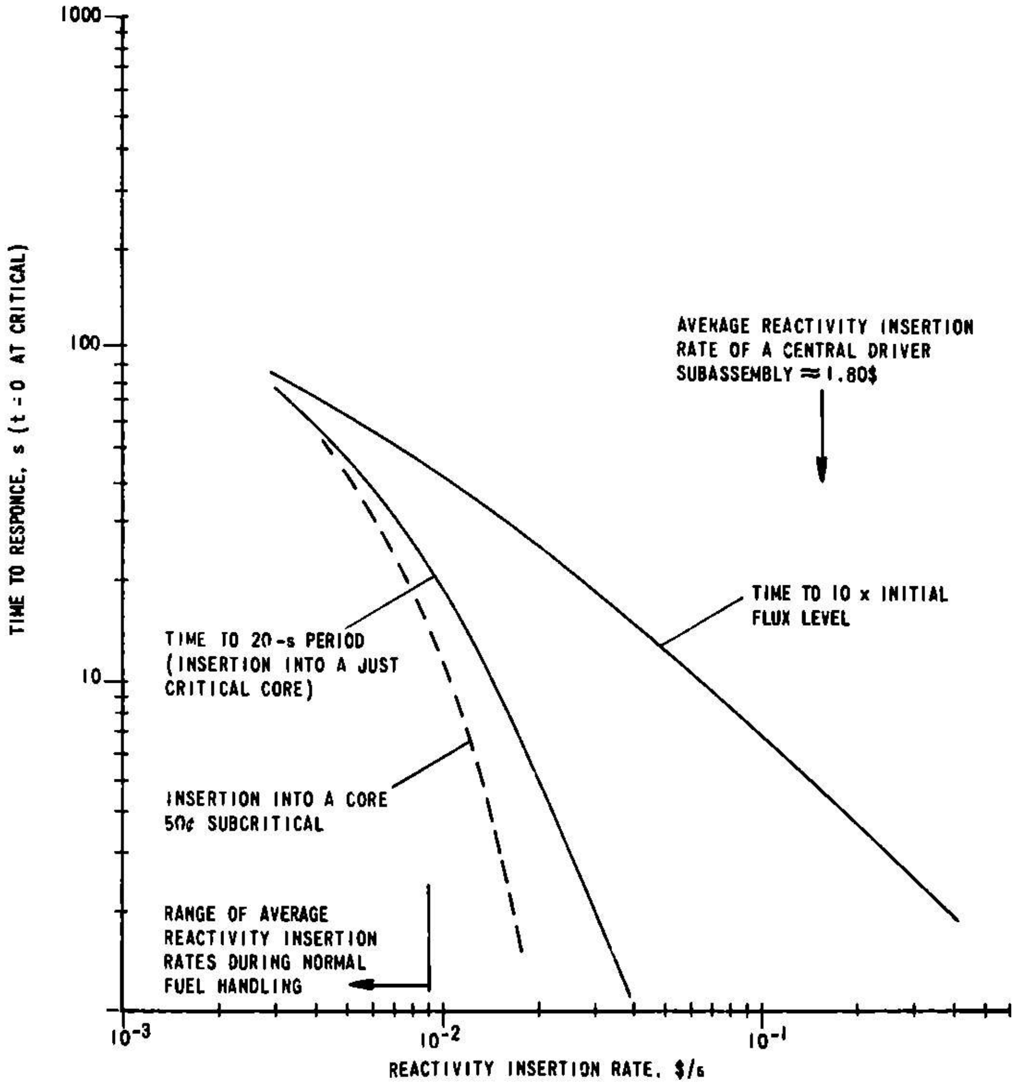


Fig. 24. System Response to a Ramp Reactivity Insertion in Fuel-handling Mode

reactivity remains to terminate any transient. EROS calculations have been made with reduced shutdown reactivity to simulate a stuck-rod condition. The following assumptions were made:

a. Two control rods are inserted at a reactor power of 62.5 W.

b. Feedback reactivity is $0.0017/MWt.

c. Control-rod drives do not have trip-assist air pressure and would have a drop time of 0.450 s for the first 10 in. (25.4 cm) of travel.

d. The highest-worth control rod, worth $1.3, is stuck, and the shutdown reactivity available is $2.4.

For a power trip at 115% of full power, the calculations show that the rods begin to move at 49.2 s and the peak in temperature of Mark-II fuel, including uncertainties, is 1199°F (648°C). For an event terminated by a period trip (17-s trip), the rods begin to move at 19.4 s and the fuel temperature remains at 700°F (~371°C).

Comparing these data with those in Table V clearly shows that the safety margin for the two-rod-insertion accident at startup, terminated by either a period or high-power trip, is not significantly changed by the revised assumptions, and adequate protection is provided. Protection is not provided by the temperature trip under the revised assumptions, but is not required from this PPS subsystem under startup conditions.

3.6.2.2 Comparison of Trips for Power Level and Subassembly Outlet Temperature. In addition to the redundancy provided by the three independent W-R channels, three diverse subsystems terminate transients with initial power below 30 MWt. The power-level trip provides the largest margin for protective actions with initial power above 24 MWt. Above 31 MWt, the period trip is no longer an effective trip device. This latter point is shown by a series of calculations for the transient initiated by insertion of two control rods at 62.5 W to 31.25 MWt.

Power-level and subassembly-outlet-temperature trips were simulated with the EROS code. Period trip was simulated with the CSMP code. Important parameters were assumed to be:

a. Feedback reactivity: \$0.0017/MWt.

b. Control-rod drop time: 0.450 s for first 10 in. (25.4 cm) of travel.

c. Worth of control rod: \$1.3.

Results of these calculations (see Table VI) show the times at which control rods begin to move and the peak cladding temperature reached, for the hottest Mark-II driver fuel, as a function of initial starting power for each trip parameter.

TABLE VI. Response of EBR-II PPS to Insertion of Two Control Rods as a Function of Initial Starting Power

	Period Trip, Setpoint 17-s Period			Power-level Trip, Setpoint 115% of Full Power		Subassembly-outlet-temp Trip, Setpoint 115% of ΔT in Rod	
Initial Power Level, MWt	Time Control Rods Move, s	Power Reached, MWt	Cladding Temp of Mark-II Fuel, °F[a]	Time Control Rods Move, s	Cladding Temp of Mark-II Fuel, °F	Time Control Rods Move, s	Cladding Temp of Mark-II Fuel, °F
62.5×10^{-6}	19.4	1.62×10^{-4}	700	49.21	1131	51.27	1624
1.0	19.51	2.60	717	38.92	1168	41.26	1382
10.0	21.22	27.4	877	31.16	1166	33.58	1290
20.0	23.20	58.4	1077	25.92	1168	28.37	1268
25.0	24.62	76.8	1198	23.74	1167	26.21	1258
30.0	26.57	99.99	1346	21.67	1168	24.11	1249
31.25	27.21	106.7	1696	21.17	1169	23.59	1248

[a]Conversion factor: °C = (°F - 32)/1.8.

Figure 25, plotted by using data from Table VI, shows the peak temperature reached in the cladding of the hottest Mark-II driver-fuel element for period trip, power trip, and subassembly-outlet-temperature trip as a function of initial starting power.

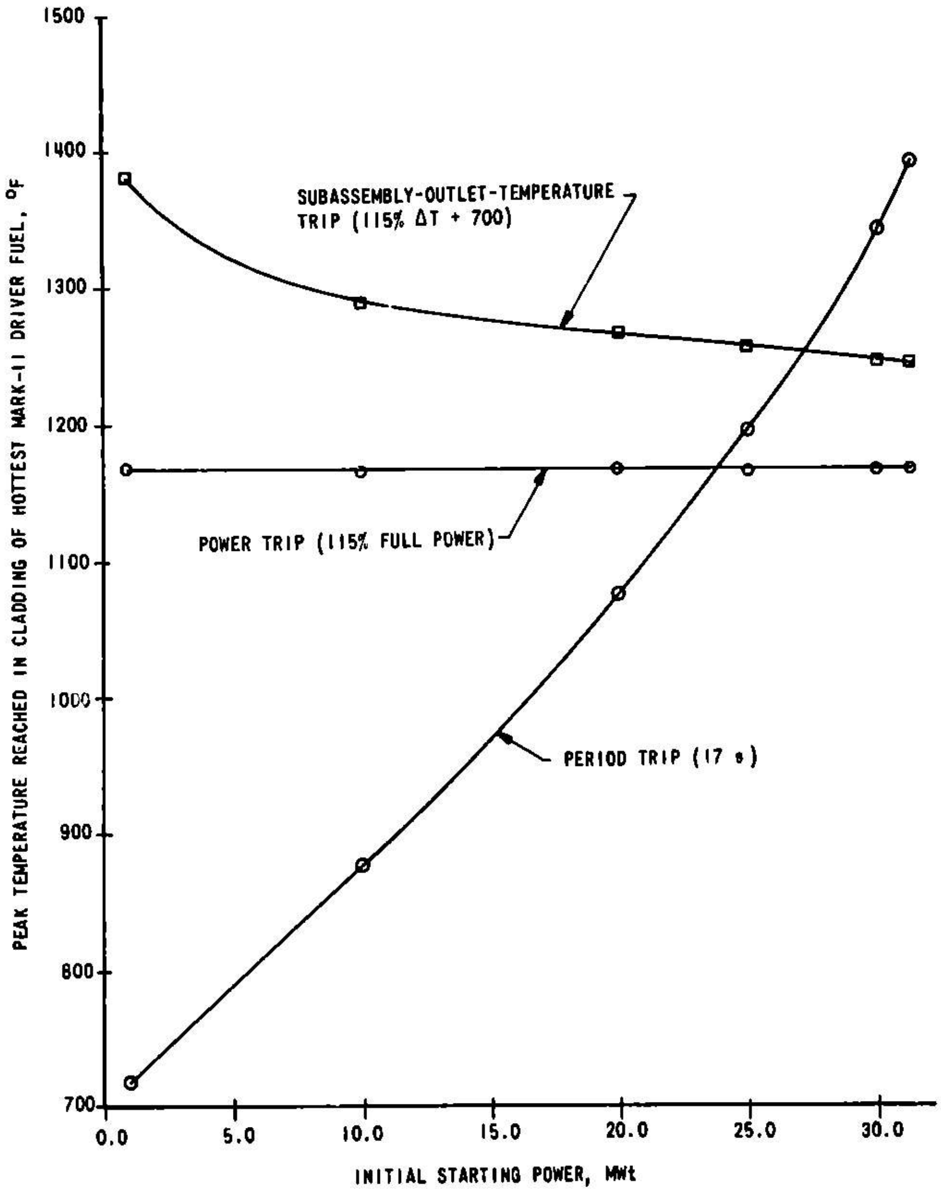


Fig. 25. Peak Temperature Reached in Cladding of Hottest Mark-II Driver-fuel Element for Period Trip, Power Trip, and Subassembly-outlet-temperature Trip as a Function of Intial Starting Power. (Simulated insertion of two control rods with feedback reactivity of 0.0017 \$/MWt.) Conversion factor: °C = (°F - 32)/1.8.

Conclusions from Fig. 25 are:

a. Period trip provides the largest protective margin from below 62.5 W to 24 MWt, with the power-level trip providing backup protection. The subassembly-outlet-temperature trip provides protection at starting powers greater than 1 MWt.

b. Above 24 MWt, power-level trips provide the largest protective margin.

c. Above 27 MWt, trips due to subassembly outlet temperature are more effective than period trips.

d. At 31 MWt, the period trip is not effective.

3.6.3 Probability of Common-mode Failures

The nine original detectors consisted of three fission chambers and six compensated ion chambers, with no uncompensated ion chambers used. The three fission chambers were identical and could be from the same manufacturer's batch. The same was true of the six compensated ion chambers. The fission chambers were functional only during fuel handling and the approach to criticality. Therefore, a common-mode failure, attributable to some unspecified common detector defect, was as credible (or incredible) with the present system as with the W-R system; in either system, only one type of detector is operational over any specific range.

In either system, all the detectors, cables, and amplifiers have not operated under the same conditions for the same period of time; thus it is difficult to postulate a mechanism for common-mode failure in such a short time that the first failure would not be detected before the second failure occurred. Even if a common-mode failure of the W-R system is considered a credible event, the subassembly-outlet-temperature channels provide diverse protections, and the reactor would be protected from damage caused by identifiable accidents with the W-R system inoperative.

3.7 Revisions to Practices and Documentation

3.7.1 Operations

The W-R instrument channels provide the same information on flux level and reactor period to the operator as the channels that were replaced. The linear level channel (Channel 7), used by the operator as the power-level control instrument, is not affected by the modification. Checkout, calibration, and maintenance procedures required changes to reflect differences between the previous instruments and the W-R channels.

Technical-specification limits have been changed to specifically identify the W-R channels where required and to specify the setpoint limit for the period-trip bypass. These limits are consistent with the requirements specified in Sec. 3.6.1.3.

3.7.2 Administrative Controls

This modification did not affect administrative controls required for safe operation of the plant.

3.7.3 Hazard Summary Report

The EBR-II Hazard Summary Report[1] describes nuclear instrumentation planned at the time the reactor was designed and placed in operation. The information affected by this modification is in the following sections of Refs. 1 and 2:

ANL-5719 (Ref. 1)	ANL-5719 Addendum[2]
Sec. III.A.5	Sec. III.A.5
Sec. IV.A.2	Sec. IV.A.1
	Sec. IV.A.2.c
	Sec. IV.A.2.e and Fig. 35

The material in Ref. 2 replaces the obsolete sections of the HSR.

3.8 Conclusion

Basic safety limits were not specifically identified in Ref. 1 or 2, which constituted the safety documentation for initial operation of EBR-II. However, the information presented in Appendixes F. and G of Ref. 2 clearly shows that failure of many fuel elements was considered as a prerequisite for a serious accident. From TREAT experiments with single EBR-II fuel elements subjected to rapid power excursions, incipient failures were observed only for peak cladding temperatures above 1742°F (950°C). The following conclusions may be reached from the information presented in Refs. 1 and 2 (see pp. 86 and 87 of Ref. 1 and pp. 7, 57, 65, and 228 of Ref. 2):

a. The temperature of formation of a fuel-cladding eutectic is a limit for sustained operation.

b. Failure from rapid transients is expected above cladding temperatures of about 1740°F (950°C).

c. No temperature limit for fuel centerline, below the point of fuel melting, has been identified for fuel-alloy performance, except as it relates to burnup performance.

d. Sodium boiling temperature was identified in a discussion of performance of fuel pins on overtemperature transients.

The EBR-II temperature limits (see Sec. 3.6.1) are consistent with or below those that may be identified in the original safety documentation.[1,2]

Another factor showing the conservatism of the limits for fuel-element temperatures is that the limits apply to the hottest element (with application of an uncertainty factor of 1.157). The average cladding temperature is about 120°F (~50°C) cooler than that of the hottest pin, and even at the stated temperature limits, only a small fraction of the core is involved. This condition is clearly more conservative than the assumption in Ref. 2 of failure of a large number of fuel elements at limiting conditions.

Data in Table F.1 of Ref. 2 summarize the protective action expected from the original nuclear-instrument channels. Comparing these data with the data in Tables II and III show that the W-R channels provide protection equivalent to that expected of the original channels for the accident cases considered in Ref. 2. Furthermore, data in Table V show that the subassembly-outlet-temperature trips provide backup protection to the nuclear-instrument channels for the operate-mode accident conditions considered in Ref. 2.

Data in Table I also show that the W-R channels are equivalent to the previous nine channels with respect to the reactor trip functions provided. In addition, data in Fig. 20 show that the W-R channels are not subject to the loss-of-overlap problem caused by high gamma-radiation backgrounds immediately after reactor shutdown.

From the above information, we conclude that the modified system is more than adequate to shut down the reactor before any basic safety limit is exceeded, under all fault conditions for which protection was claimed in either of the original safety documents.[1,2] In several respects, the modified system is superior to the nine-channel system being replaced.

4. EBR-II PLANT MODIFICATION NO. 443: UPGRADING LOSS-OF-FLOW PROTECTION

J. F. Boland, R. H. Curran, E. M. Dean, and V. N. Thompson

4.1 Summary

The subsystem for loss-of-flow (LOF) protection of the EBR-II PPS must be able to shut down the reactor, after any LOF event that could reasonably be expected during the plant lifetime, before the primary or secondary containment systems would be challenged. Additional requirements for this protection subsystem, related to the protection of the plant from damage, are in RDT C16-1T.[4] The plant-damage limitations place more severe design requirements on the instrument system, and these requirements have set system-design criteria.

The primary LOF protection subsystem consists of four flowmeter channels connected in one-out-of-four logic; a diverse secondary LOF protection subsystem consists of four channels for subassembly-outlet temperature connected in two-out-of-four logic. Analyses in this section show that the primary or secondary subsystem each will prevent limits from being reached for all anticipated and unlikely accidents, and some extremely unlikely accidents.

Based on the above conclusions, PM 443 removed many anticipatory LOF trips originally included in the reactor shutdown system when the plant was designed and no data were available on the actual performance of the flowmeter or subassembly-outlet-temperature channels. One anticipatory LOF trip, for undervoltage of the 2400-V bus, is retained to keep desirable existing reliability and diversity for the anticipated fault of loss of primary pumping power, which can be expected about twice per year. With this fault, reactor shutdown, before coolant flow has measurably decreased, is desirable to minimize temperature transients in the driver fuel and experiments.

This modification did not introduce any new safety hazards, and the modified system provides protection against any LOF accident for which the previous system provided protection. Removal of the anticipatory trips simplifies the shutdown system and reduces the susceptibility of the plant to spurious trips that have subjected the cooling systems and experiments to undesirable thermal transients.

This modification was completed in April 1975.

4.2 System before Modification

The premodification LOF protective subsystem consisted of four low-flow trips in one-out-of-four trip logic, four subassembly-outlet-temperature trips in two-out-of-four trip logic, and 11 anticipatory pump trips.

4.3 System after Modification

The affected sections of the shutdown system before and after the modification are shown in Fig. 26. The modified system does not contain relay contacts for the deleted trip functions, and separate relay contacts are provided for each trip that responds to low coolant flow in the low-pressure plenum.

4.3.1 Alarm Indications

Each trip function is connected to an alarm point on the shutdown section of the console annunciator panel. Each channel of the subsystem for subassembly outlet temperature is connected to an alarm point on the non-shutdown section of the console annunciator panel to indicate when one of the four channels has tripped.

The alarms to indicate abnormal conditions in the primary-pump system are on the annunciators for primary pumps No. 1 and 2 in the corridor panel containing the readout instrumentation for parameters of the pump systems. An alarm for each pump is located in the control room, on the console annunciator panel.

4.3.2 Instrument Channels

4.3.2.1 Total Reactor Flow. Figure 27 is a schematic diagram for this instrument channel. Modifications to this channel are:

a. Installation of buffer-amplifiers to isolate non-PPS portions of the channel-output circuits from PPS circuits.

b. Installation of a millivolt test source, to introduce a test voltage in place of the transducer signal for channel calibration, and interlock circuits to activate the trip relay when the test source is switched into the circuit.

c. Mounting components on new chassis.

For this channel and those described in Secs. 4.3.2.2, 4.3.2.3, 4.3.2.4, and 4.3.2.5, the MV/I converters, input buffer-amplifier, trip units, and relays in the PPS portions of the modified channel are identical to those in the unmodified system. Provision has been made for reducing the time constants of noise filters to improve the time response of the channel if "as-built" data show that sufficient noise rejection can be obtained with less filtering. Therefore, basic precision, time response, and reliability of the modified channel are equal to, or better than, those of the unmodified channel.

4.3.2.2 Reactor Outlet Temperature. This channel does not provide a signal to the shutdown system but provides temperature compensation to the instrument for total reactor coolant flow. Therefore, the portion of this channel

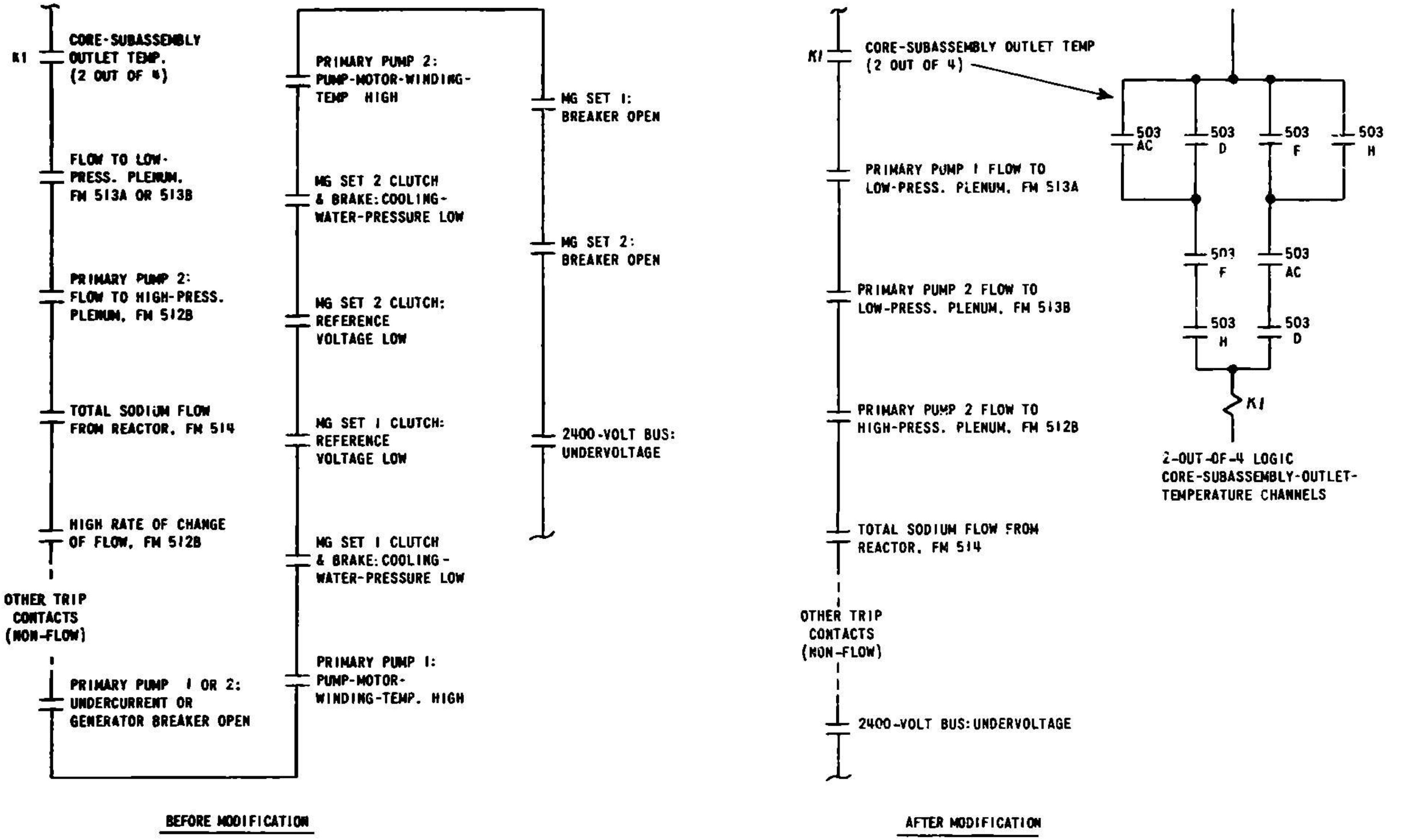


Fig. 26. Partial Diagram of Shutdown Circuit, Showing Flow-related Trip Contacts before and after Completion of PM 443

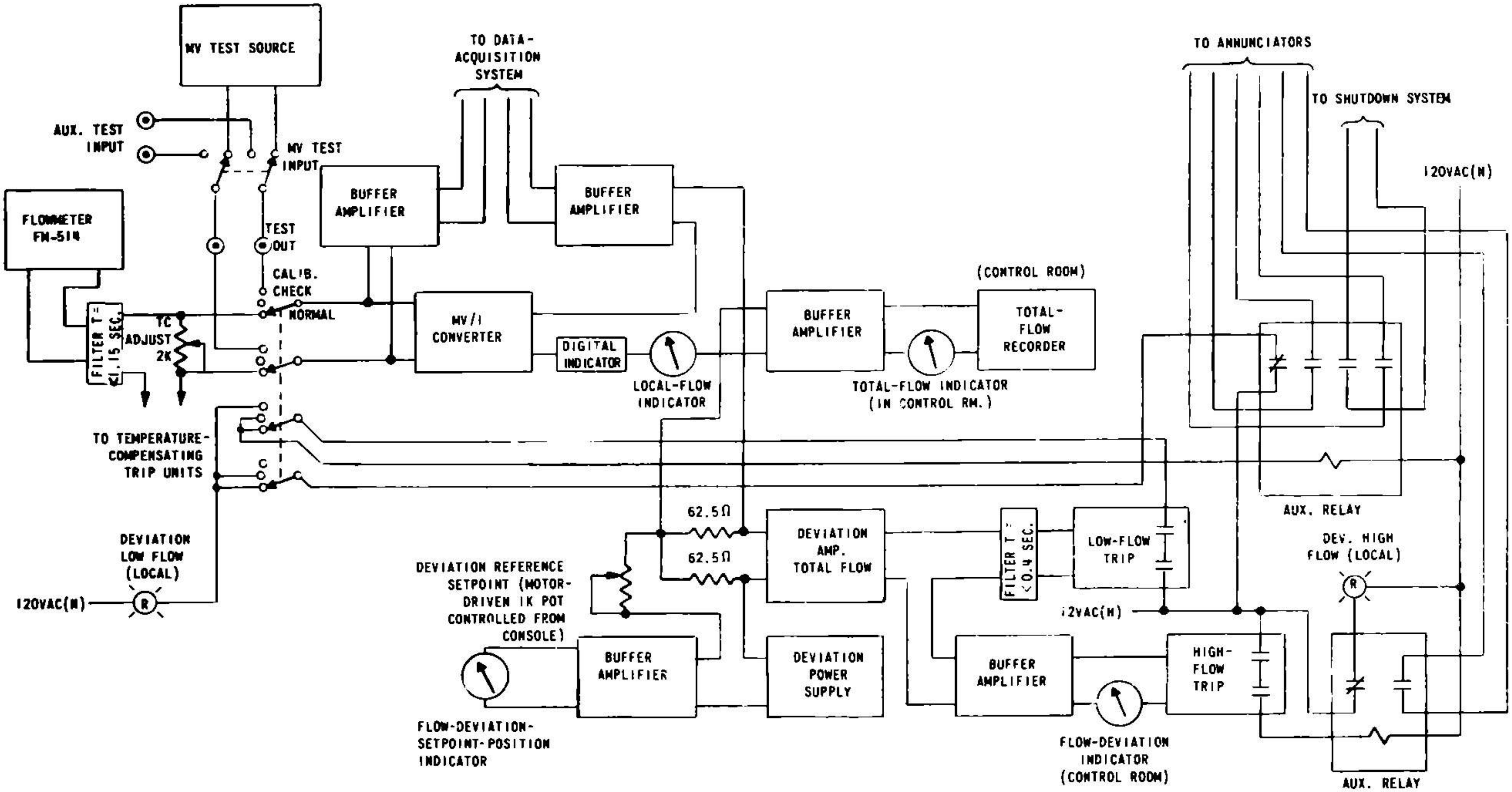


Fig. 27. Schematic Drawing of Channel for Total Reactor–coolant Flow

required for temperature compensation is considered part of the PPS. Figure 28 is a schematic diagram for this instrument channel. Modifications to this channel made by PM 443 are the same as those listed in Sec. 4.3.2.1, except that under item b, the interlocking circuits provide an alarm when the test source is switched into the system.

4.3.2.3 Coolant Flow in High-pressure Plenum. Figure 29 is a schematic diagram for this instrument channel. The modifications made to this channel are the same as those listed in Sec. 4.3.2.1. Also, the circuit for rate-of-change trip, alarm, and indication is removed.

The rate-of-change trip was removed as an unnecessary anticipatory trip, and the associated alarm and indication functions were removed because they would not provide useful information to the operators.

4.3.2.4 Coolant Flow in Low-pressure Plenum. Figure 30 is a schematic diagram for these instrument channels. Modifications made to these instrument channels are the same as those listed in Sec. 4.3.2.1.

4.3.2.5 Subassembly Outlet Temperature. Figure 31 is a schematic diagram for these instrument channels. Modifications made to these channels are the same as items b and c in Sec. 4.3.2.1.

4.3.2.6 Loss of Pumping Power. The trip circuit for loss of power to the 2400-V bus is not modified by PM 443.

4.3.3 Equipment Arrangement

The components for each flow channel, each subassembly-outlet-temperature channel, and the reactor-outlet-temperature channel are housed in a separate chassis. The new chassis and 17 reused chassis are arranged in six instrument cabinets (see Fig. 32). The wiring for PPS and non-PPS circuits are separated to the maximum extent practicable within the cabinets; the wiring is run in separate cables from the six new cabinets to other cabinets and components in the plant.

The instrument chassis for each redundant channel of a protective subsystem is housed in a panel separate from every other redundant channel of that subsystem. The trip relay for each channel of a subsystem is in the panel with the associated instrument chassis; thus isolation and physical separation are provided. The sensor leads to the four flowmeters are in separate cables in a common conduit except for a short distance in the reactor building, where it was considered impracticable to make a modification to provide separate conduits. The subassembly outlet-temperature sensors are also in a common conduit where they pass from the reactor; providing physical separation at this point is impossible.

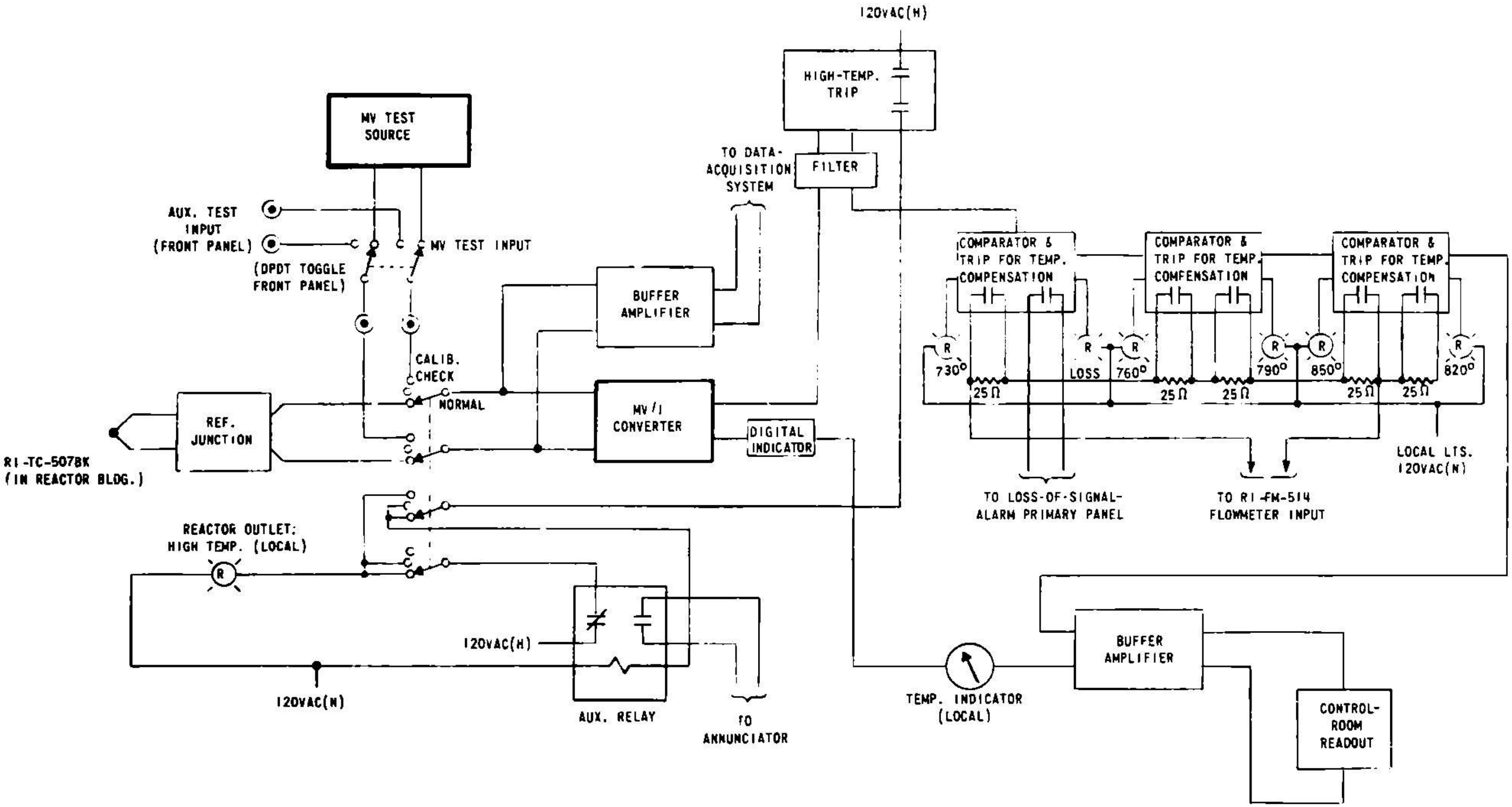


Fig. 28. Schematic Drawing of Channel for Reactor Outlet Temperature

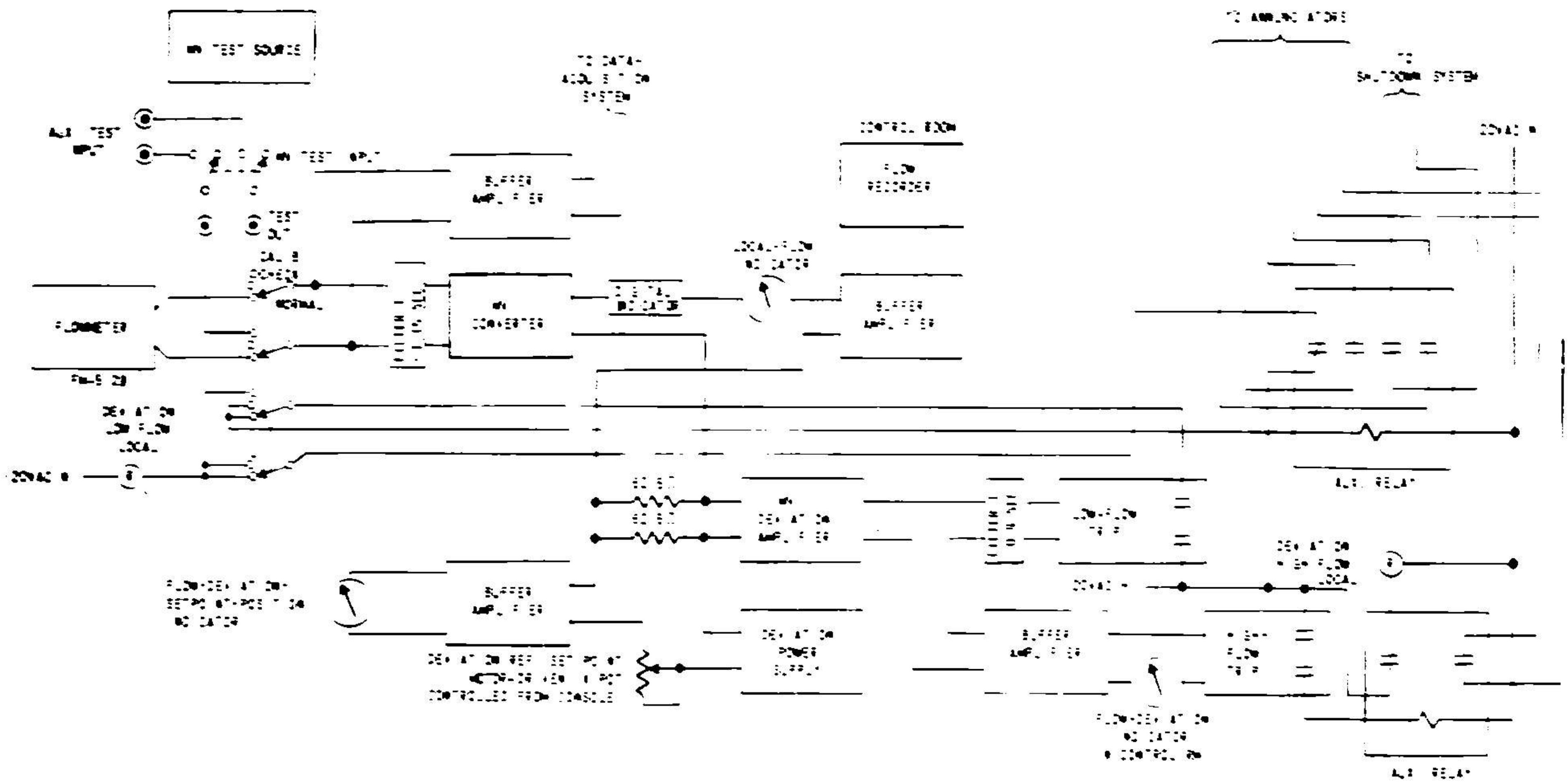
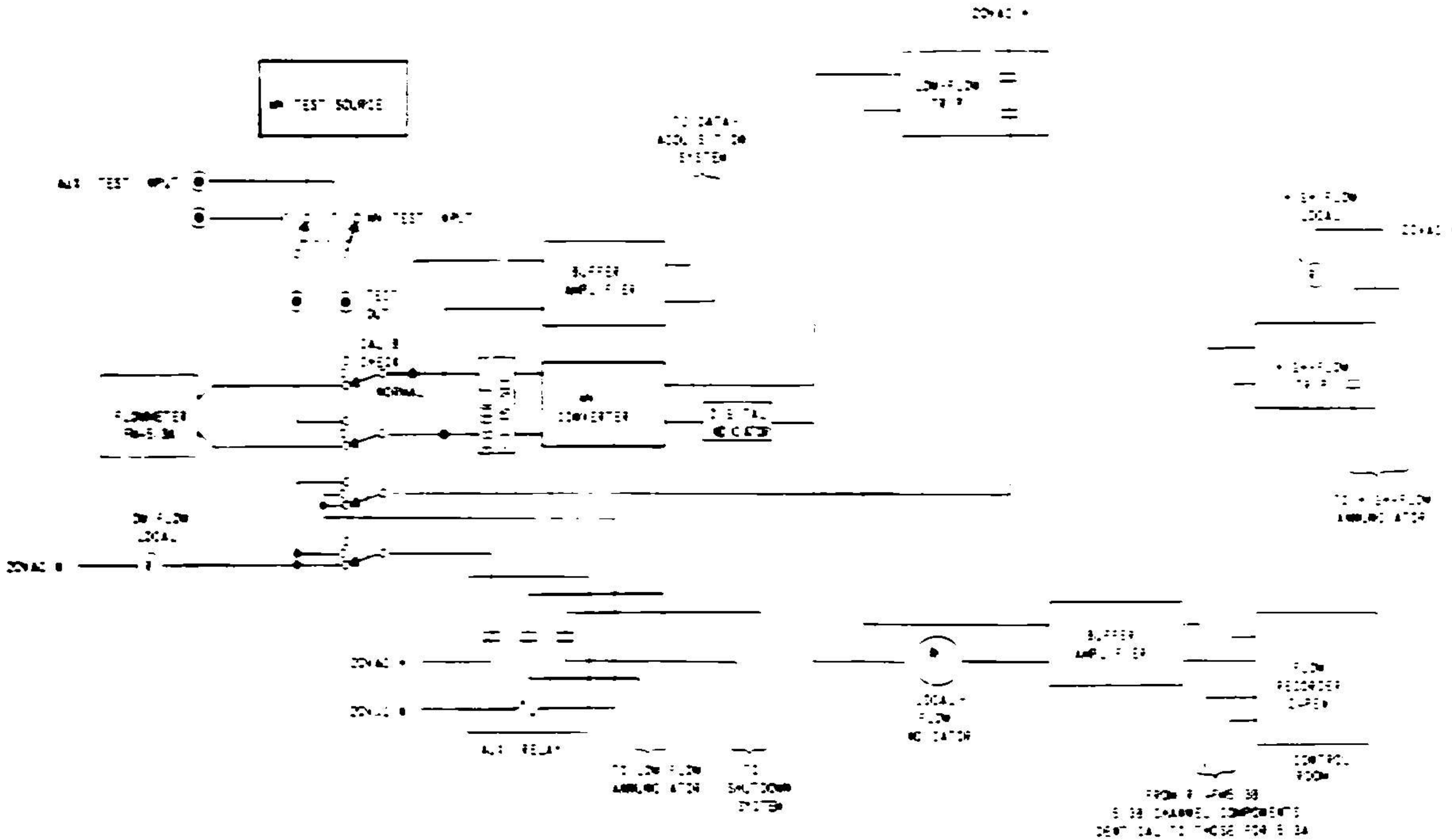


Fig. 24. Schematic Drawing of Flow in High-Pressure Portion of Pump No. 2


mA TEST SOURCE
NORMAL
BUFFER AMPLIFIER
CONVERTER

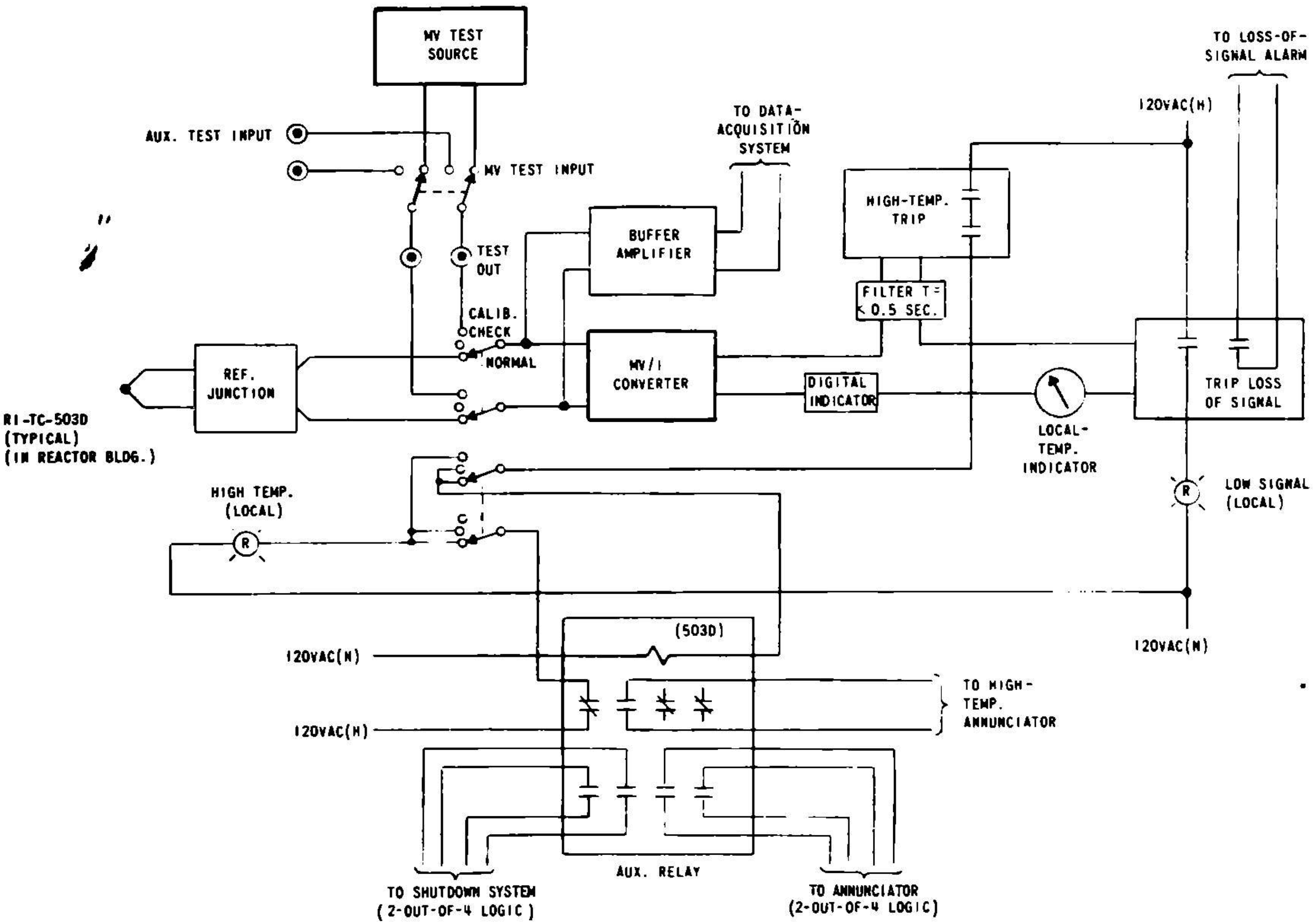


Fig. 31. Schematic Drawing of Channels for Subassembly Outlet Temperature

18	19	20	21	22	23
CORE SUBASSEMBLY-OUTLET TEMP. RI-TC-503D	NUCLEAR CHANNELS 9, 10, & 11	CORE SUBASSEMBLY-OUTLET TEMP. RI-TC-503F	CORE SUBASSEMBLY-OUTLET TEMP. RI-TC-503H	BULK-Na & REACTOR-OUTLET-PRESSURE ALARM UNITS	CORE SUBASSEMBLY-OUTLET TEMP. RI-TC-503AC
PUMP #2 HIGH PRESSURE FM-512B		BLANK PANEL	RELAYS FOR REMOTE READOUT	POWER SUPPLY & RELAY CHASSIS FOR BULK Na	PUMP #2 LOW-PRESS.-PLENUM FLOW CONVERTER & TRIP FOR FM-513-B
PUMP #2 HIGH PRESSURE FM-512B		REACTOR OUTLET TEMP. RI-TC-507BK	BLANK PANEL TO PERMIT ACCESS TO PANEL BELOW	BULK-SODIUM TEMP. INDICATORS & TEST EQUIPMENT	THIMBLE-COOLING TEMP. CHANNEL
PUMP #1 RPM CONVERTER & TEST SOURCE		REACTOR OUTLET TEMP. THERMOCOUPLE RI-TC-507BK ELECT. COMP. FOR TEMP. COMP. OF TOTAL FLOW FM-514	TOTAL-FLOW CONVERTER TEST SOURCE & BUFFER AMPLIFIERS FOR FM-514	BULK-SODIUM TEMP. CONVERTERS & TRIPS 540AR 540AS	THIMBLE-COOLING TEMP. CHANNELS TEST EQUIP.
PUMP #1 RPM DEVIATION AMP. POWER SUPPLY HIGH & LOW INDICATOR TRIPS		SECONDARY-SODIUM FLOW AMPLIFIER & ALARM COMPONENTS	TOTAL-FLOW DEVIATION AMPLIFIER. POWER SUPPLY, TRIP CIRCUITS & BUFFER AMP. FOR FM-514	BULK-SODIUM TEMP. CONVERTERS & TRIPS 540AT 540AV	THIMBLE-COOLING TEMP. CHANNEL
				NEWPORT BUFFERS & POWER SUPPLIES	THIMBLE-COOLING TEMP. CHANNEL
				PUMP #1 LOW PRESSURE FM-513A	

Fig. 32. EBR-II Process-control Racks 18-23: Panel Layout for PM 443

4.3.4 Conversion of Anticipatory Trips to Alarms

The abnormal-condition trips listed in Table VII are removed from the EBR-II shutdown string by PM 443, but are retained as alarms; the annunciator windows used on the corridor panel are also used here.

TABLE VII. Anticipatory Trips Converted to Alarms by PM 443

Previous Shutdown-circuit Contacts	Abnormal Condition
174 CP1-3	Generators No. 1 or 2: output breaker open--or primary pumps No. 1 or 2: low current to pump
CR2	Primary pump No. 1 or 2: high temperature of pump-motor winding
WP	Motor-generator set No. 1 or 2: low cooling-water pressure for clutch and brake
IS	Motor-generator set No. 1 or 2: low coupling-clutch voltage
52A	2400-V supply voltage breaker opened to motor-generator set No. 1 or 2

4.3.5 Characteristics of Instrument Channels

Performance characteristics for the EBR-II flowmeter and subassembly-outlet-temperature channels required to prevent safety limits from being exceeded during LOF transients were determined from the EROS calculations discussed in Appendix A. Essential performance requirements are based on RDT Standard C16-1T[4] and defined in Ref. 10. Step-function response characteristics of the instrument channels, exclusive of the sensing elements, are readily measurable and can be calculated from the transfer-function equations of the instrument channel. (See Appendix A, Sec. 2.3.) The time-response requirements given in Table VII for the instrument channels are for step-function signals applied at the input to the measuring circuit. Maximum time-response values are given with the trip points set at the limiting values (see Sec. 4.6.3), and again for nominal values. (These values do not include the response of the sensing elements or control-rod-drive mechanisms.)

Instrument channels are tested with the trip point at the nominal value and must meet the time-response requirements given in Table VIII. Flowmeter response time may be neglected;[11] the thermocouples have a measured time constant of less than 0.5 s. If the requirements for step-function response in Table VIII are met and the sensing elements and rod mechanisms are operating normally, meeting the system-response requirements specified for EBR-II is ensured. These time-response characteristics are applicable to both the unmodified and modified systems.

4.3.5.1 Precision. The precision of the instrument channels has been calculated[10] (see Table VIII). The normal setpoint for each instrument channel must be selected to obtain the required trip point if the setpoint changes by an amount equal to the precision limit for the channel. The trip points used in the EROS calculations discussed in Appendix A and the limiting safety-system settings for EBR-II were selected to allow an adequate margin between normal setpoints and limiting trip points to account for the precision of the channels. The precision is unchanged by PM 443.

TABLE VIII. Performance Characteristics of LOF Protection Channels

			Step-function Responses of Instrument Channels				
			Trip-point Limiting Values		Trip-point Nominal Values		
Channel	Failures per Hour	Calculated Instrument Precision	Trip Point	Required Time Response,[a] s	Trip Point	Required Time Response,[a] s	Driving Function
Total flow	6.6×10^{-5}	1.1%	88%	1.12	94%	0.92	Step 100 to 0.0%
HPP flow	6.6×10^{-5}	1.1%	88%	1.12	94%	0.92	Step 100 to 0.0%
LPP flow	4.1×10^{-5}	1.5%	85%	0.65	87%	0.62	Step 100 to 0.0%
Subassembly outlet temperature	4.6×10^{-5}	4.6°F	847°F (-453°C)	1.29	840°F (-449°C)	1.22	Step 700°F (-371°C) to 900°F (-482°C)

[a]Time from injection of the simulated signal to the time that the trip relay is deenergized. Control-rod motion would occur 0.020 s later. (See Table XV, Appendix A.)

4.3.5.2 Reliability. The failure rates for the components in the flowmeter and subassembly-outlet-temperature channels have been determined[10] from maintenance records at EBR-II; the channel failure rates listed in Table VII were calculated as the sum of the failure rates of the components. The calculated failure rates include both safe and unsafe failures; the rate for unsafe failures would be considerably less.

4.3.6 Reliability of Subsystem for LOF Protection

The reliability of this subsystem, without the loss-of-power trip, has been determined from the rates of single-channel failures and the system configuration. The flowmeter-protection subsystem has a failure probability of 7.3×10^{-6} for double-pump faults and 2.7×10^{-3} for single-pump faults during a normal reactor run of 1000 h. The protection subsystem for subassembly outlet temperature has a failure probability of 3.8×10^{-4} for a 1000-h run.

These values are for any type of failure. Each instrument channel is monitored by the operators for proper operation, and the probability of a channel failure being undetected for more than a few hours is low. Therefore, an upper limit for the probability of a total failure of the LOF protection subsystem during a 1000-h reactor run is 10^{-6}.

4.3.7 Environmental Conditions for Instrument Channels

4.3.7.1 Instrument Cabinets. The instrument cabinets are in the cable-routing room and have conditioned cooling air. Each cabinet has a high-temperature alarm set at 115°F (~47°C). These environmental conditions are unchanged by PM 443.

4.3.7.2 Flowmeters. The magnetic flowmeters are in the primary system and are subject to the same environment as the piping system in which they are installed. The flowmeters are designed to operate up to the demagnetization temperature of 1450°F (~788°C). The environmental conditions for the flowmeters are unchanged by PM 443.

4.3.7.3 Thermocouples. The subassembly-outlet-temperature sensors are Chromel-Alumel thermocouples subject to the environment of the reactor outlet plenum. They are designed to operate up to 1650°F (~900°C). Their environmental conditions are unchanged by PM 443.

4.3.8 Power Requirements

The instrument channels are fail-safe on loss of electrical power, and no power is required for the protection actions to be completed after any LOF event. This feature is unchanged by PM 443.

4.4 Justification for Modification

This modification upgrades the LOF protection subsystem by:

a. Reducing the number of trip contacts in the PPS.

b. Reducing the susceptibility of the plant to spurious trips.

c. Providing a subsystem that meets the requirements of RDT C16-1T for LOF protection insofar as practicable.

The 10 anticipatory types of trips removed by PM 443 were originally included in the shutdown system when the plant was designed for use as a reactor experiment, and data were not available on the performance characteristics of the instrument channels or the reactor core. Retention of these trips in the shutdown system of the reactor, as now used as an irradiation facility, is undesirable because spurious trips subject the reactor and its experiments to unnecessary thermal transients.

The arrangement of redundant instrument channels in the unmodified system made them unnecessarily susceptible to failures from a single event, such as a fire in a single instrument cabinet. The modified system reduces the susceptibility of the system to common-mode failures.

4.5 Applicable Standards

This modification meets the requirements of the following standard except for the specified variances described.

Number	Title
RDT C16-1T	Supplementary Criteria and Requirements for RDT Reactor Plant Protection Systems[4]

The flowmeter subsystem does not meet the coincidence recommendation of Sec. 4.2.5 of RDT C16-1T because only two flowmeters are available to detect failure of primary pump No. 1; with this restriction, a coincidence arrangement is not possible.

Physical separation and isolation of the four redundant flow channels and four redundant thermocouple channels are considered to conform with RDT C16-1T, with two exceptions: The sensor leads to the flowmeters and the sensors for subassembly outlet temperatures are in a common conduit for a short length (see Sec. 4.3.3). These wiring variances are not considered significant, because the wiring for the primary and backup protection subsystems is not subject to failure from any single credible event.

The 2400-V undervoltage trip is an anticipatory trip and its inclusion in the PPS requires justification. (See Sec. 3.5 of RDT C16-1T.[4]) Power loss to both primary coolant pumps has often occurred and is expected once or twice per year. The flowmeters and channels for subassembly outlet temperature require a measurable reduction in flow or increase in temperature to cause a reactor trip, a small increase in fuel and coolant temperatures, above those extant during normal operation, will occur before the reactor is shut down by these trips after a loss of power. The undervoltage trip starts action before the flow has decreased measurably and thereby minimizes the "first-peak" temperatures. Reducing these anticipated temperature transients to minimal practicable values is desirable, and the PPS is the only available system capable of doing so.

The 2400-V undervoltage trip does not meet the coincidence requirements of Sec. 4.2.5 of RDT C16-1T,[4] but this circuit has not failed in 10 years of operation. The failure rate of this single channel is considered low enough that coincidence is not required to prevent random failures from unduly reducing plant availability.

4.6 Safety Analysis

4.6.1 Safety Limits Related to LOF Events

The safety limits developed for EBR-II technical specifications and applied to loss-of-flow (LOF) events relate to the performance of fuel-element cladding at high temperature. The limits have been developed to ensure that loss of cladding integrity is of low probability and will not constitute a safety hazard. Those limits are:

a. Temperatures for formation of fuel-cladding eutectic will not be exceeded for more than 60 s.

b. Fuel-cladding interface temperatures will not exceed 1500°F (~816°C).

These limits ensure that cladding loss would be limited to less than 5% of nominal thickness and are below temperatures at which tests[12-14] have indicated fuel failure.

4.6.2 Classification of LOF Events

The LOF events identified in Table IX have been identified and analyzed to define the limiting case for establishing response-time requirements of PPS instrumentation. As discussed in Sec. 4.6.3, trip-setpoint requirements are not established by LOF-transient cases. The limiting case (i.e., the most rapid flow transient) was determined to be that of single-pump seizure with coastdown of the second pump. The other events, related to loss of pumping power, do not result in a more rapid rate of coastdown than the pump-seizure case (see Fig. 33). The results of analysis for that seizure case are presented in Appendix A and discussed in Sec. 4.6.3.

TABLE IX. Categorized LOF-related System Faults Related to First-peak Temperatures That Determine Design Criteria for PPS Instrumentation Initiating Protective Function

Category	System Fault
Anticipated	Loss of primary pumping power
	Loss of power to one pump
Unlikely	Seizure of one primary pump
	Loss of primary pumping power and failure of all flow-level trips
	Seizure of one primary pump and simultaneous coastdown of the second primary pump
	Seizure of one primary pump and failure of flow-level trips
	Loss of primary pumping power due to simultaneous opening of pump circuit breakers
Extremely unlikely	Seizure of one primary pump, coastdown of the second pump, and failure of all flow-level trips
	Any failure of pumps or pumping power with failure of both flow and temperature trips

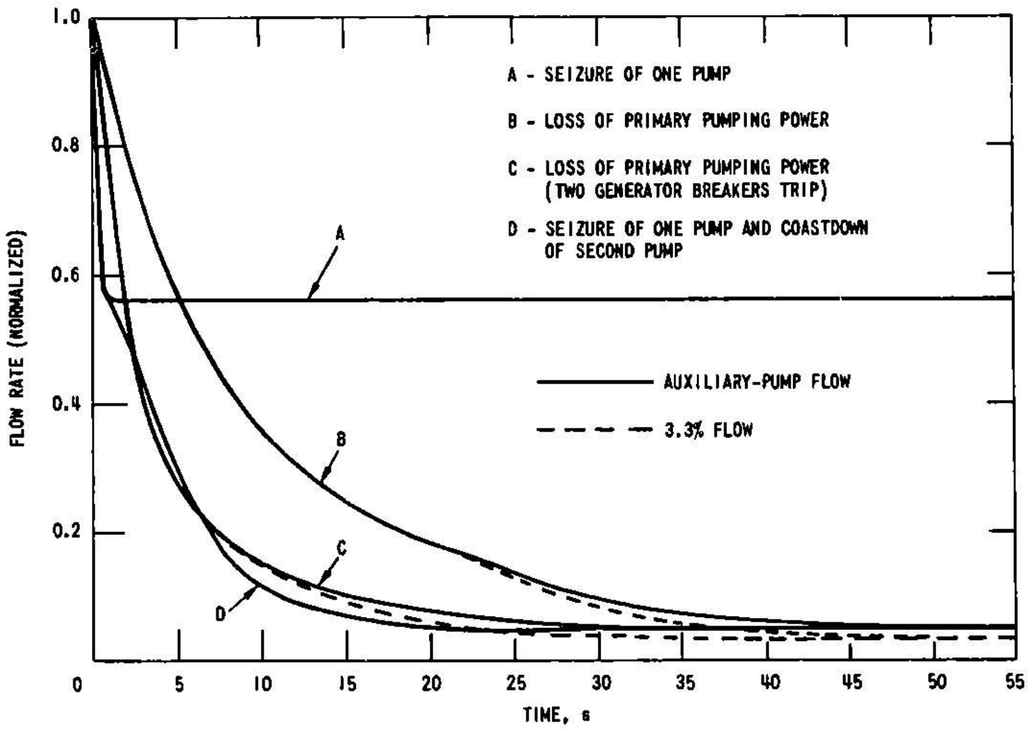


Fig. 33. Flow–coastdown Data for Safety Analysis

LOF cases may be hypothesized in which malfunctions follow the initial fault (for example, sticking of a control rod or failure of one or more of the protective trip channels), but these events do not affect the response-time or setpoint requirements of PPS instrumentation. Reliability characteristics of the PPS are discussed in Sec. 4.3.6.

4.6.3 Performance Requirements for Shutdown System

Limiting setpoint requirements for trips for flow and subassembly outlet temperature have been established as follows:

a. Trip on total flow and high-pressure-plenum flow: 88% of full reactor flow.

b. Trip on low-pressure-plenum flow: 85% of full reactor flow.

c. Selected* trips for subassembly outlet temperature: 115% of measured rise in subassembly coolant temperature (above the reactor inlet temperature) occurring at full reactor power, not to exceed 35°F (~19°C) above the measured subassembly outlet temperature.

Those limiting setpoint requirements are established for off-normal flow conditions that may be hypothesized to occur slowly (for example, from operator error in adjustment of flow). They ensure that the lower temperature

*Selected thermocouples for trips due to subassembly outlet temperature must meet the following criteria: (a) The thermocouple must be above a fueled subassembly, (b) the subassembly ΔT at full power must be 120°F (~67°C) or greater.

bound--formation of fuel-cladding eutectic--will not be exceeded and the 60-s time limit above this temperature, therefore, not be a factor. (Temperature for eutectic formation requires at least 31% increase above the nominal temperature rise.)

Time-response requirements of PPS instrumentation are determined from the hypothesized faults resulting in rapid temperature transients. For these, two safety-limit bounds are applied:

a. Eutectic temperatures must not be exceeded for greater than 60 s.

b. Peak temperatures must not exceed 1500°F (~816°C).

Figure 34 shows the temperature response of the system to three LOF faults. The limiting case for establishing required response time is Case B: single-pump seizure and coastdown of the second pump. For this case, 3.21 s are available for protective action before the safety limit is exceeded. Because of the slower temperature increase for slower loss of flow (for example, Case A), the limiting response times established by the seizure case are short enough to prevent even eutectic temperatures from being exceeded for the anticipated cases of loss of primary pumping power. The combination of setpoint and response-time requirements as established by the respective limiting cases (pump seizure and operator control error) therefore ensure protection for the full range of intermediate cases.

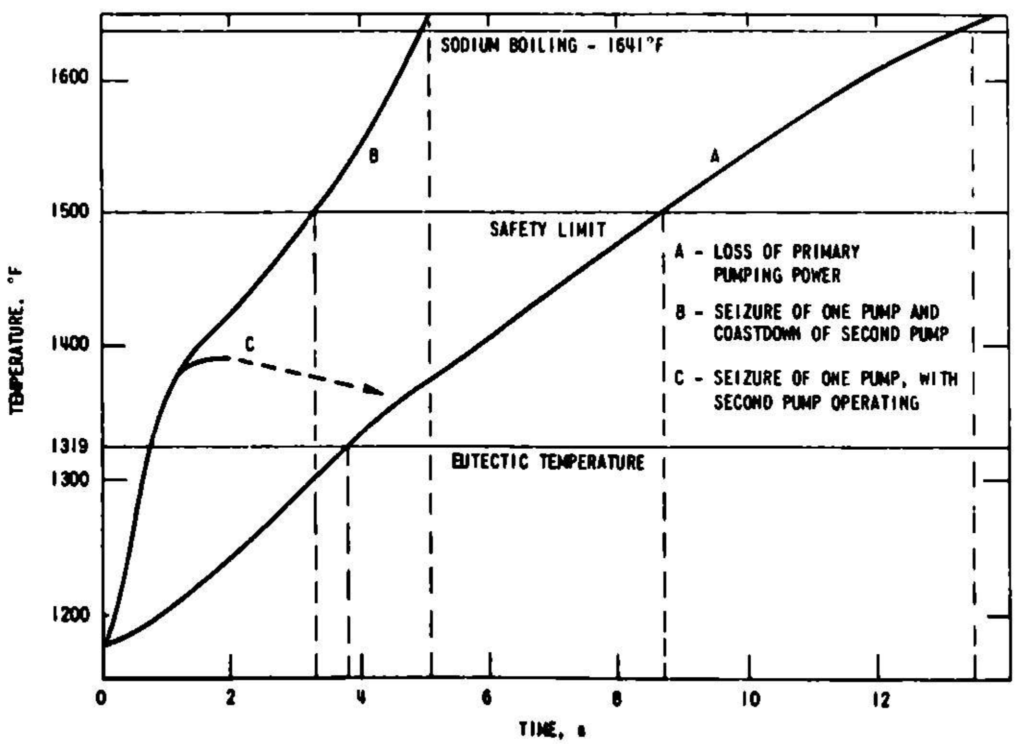


Fig. 34. Temperature as a Function of Time (with Uncertainties) for Hottest Mark-II Driver Fuel in Row 6 after Loss of Coolant Flow. Conversion factor: °C = (°F - 32)/1.8.

4.7 Revisions to Practices and Documentation

4.7.1 Operations

Operating instructions now instruct the operator to take corrective action when a "pump abnormal" alarm is received and initiate an anticipatory reactor shutdown if the problem cannot be corrected. These instructions formerly covered pump-alarm conditions that did not cause a reactor shutdown, and the same instructions apply to the trips that are converted to alarms by PM 443.

Operating limits were changed to reflect the removal of the rate-of-change-of-flow trip for pump No. 2.

4.7.2 Administrative Controls

Administrative controls required for safe operation of the plant are not affected by PM 443.

4.7.3 Hazard Summary Report

The following sections of Refs. 1 and 2 are superseded:

ANL-5719 (Ref. 1)	ANL-5719 Addendum[2]
Table VII	Table III
	Fig. 35
	Table K-1

4.8 Conclusion

Basic safety limits were not specifically identified in Ref. 1 or 2 that constituted the safety documentation for initial operation of EBR-II. However, the information in Appendixes F and G of Ref. 2 clearly shows that failure of a large number of fuel elements was considered a prerequisite for a serious accident. From TREAT experiments with single EBR-II fuel elements subjected to rapid power excursions, incipient failures were observed only for peak cladding temperatures above 1740°F (~968°C). The following conclusions may be reached from the information in Ref. 1 (pp. 86 and 87) and Ref 2 (pp. 7, 57, 65, 259, 228, and 239):

a. Temperature of formation of fuel-cladding eutectic is a limit for sustained operation.

b. Incipient failure from rapid transients is expected to occur above cladding temperatures of about 1740°F (~967°C).

c. No temperature limit for fuel centerline, below the point of fuel melting, has been identified for fuel-alloy performance, except as it relates to burnup performance.

d. Sodium-boiling temperature was identified in a discussion of performance of fuel pins on over-temperature transients.

The temperature limits established (see Sec. 4.6.1) are consistent with/or below those in Refs. 1 and 2.

Another factor in showing the conservatism of the new limits on fuel-element temperature is that the limits apply to the hottest element (with application of an uncertainty factor of 1.157). The average pin-cladding temperature is about 120°F (~67°C) below the highest temperatures, and even at the stated temperature limits, only a small fraction of the core is involved. (Figure 48 in Appendix A shows the fraction of core exceeding the temperature of eutectic formation as a function of time for an unprotected loss of primary pumping power.) This is clearly more conservative than the HSR assumption of failure of many fuel elements at limiting conditions.

Data in Figs. F-2 through F-5 of Ref. 2 were for evaluation of the shutdown reactivity required to prevent excessive temperature after a transient due to loss of pumping power and were based on the assumption that the shutdown system would operate fast enough to prevent excessive temperatures during flow reduction before control rods were inserted. Obviously, the shutdown-delay times assumed in these calculations were unrealistic for flow-sensing instruments because the flow would not reduce measurably (see Fig. F-1 of Ref. 2) in 0.2 s. Trip-time requirements for shutdown time for the various trip channels were not given in Ref. 2; thus the response-time requirements for the modified shutdown system cannot be compared with that described in Ref. 2. Because the undervoltage-trip circuit and the flowmeter-trip circuits are not removed by this modification, the statement "The trip signal is derived either from sensing loss of electrical power to one or more pumps or from sensing the abnormal reduction in primary system flow rate" on p. 214 of Ref. 1 is applicable to the modified system.

In any case, data in Table XVI, Appendix A, show that the temperatures reached, for all the anticipated and unlikely faults considered, before the control rods would have been inserted, are well below the safety limit of 1500°F (~816°C). Furthermore, the trips for both flow and for subassembly outlet temperature prevent the temperatures of the fuel-cladding interface from exceeding the temperature of eutectic formation before shutdown, with the loss-of-pumping-power conditions given in Ref. 2. This conclusion applies for the worst-case trip point and the time-response conditions applicable to the modified system. (See Table XVI of Appendix A.) Therefore, the modified system is more than adequate to shut down the reactor before the basic safety limit is exceeded, under all fault conditions for which protection was claimed in either Ref. 1 or 2.

Thus installation of PM 443 does not adversely affect the safety of the EBR-II reactor plant during normal operations or under any accident conditions discussed in Ref. 1 or previous addenda.

5. EBR-II PLANT MODIFICATION NO. 404: REMOVAL OF THE CONTROL-ROD-NOT-LATCHED TRIP CONTACTS FROM THE REACTOR SHUTDOWN SYSTEM

V. N. Thompson

5.1 Summary

Before Plant Modification 404, the EBR-II reactor shutdown system contained trip contacts that opened to automatically trip the reactor if any control rod became unlatched from its control-rod drive. This modification changed the system so an automatic trip would not occur when a control rod unlatched. The condition of all-control-rods-latched is incorporated in the system as a startup "permissive" condition required only to initially energize the system. In addition to preventing reactor startup when any control rod is unlatched, the contacts for control-rod-not-latched prevent resetting the shutdown system after a trip until all control-rod drives are driven to the down position.

A control-room alarm will be received when any control rod unlatches. Also, the lights on the nuclear panel that indicate LATCHED status for each control rod and the present rod-latch input to the data acquisition system (DAS) were not changed by this modification. This modification was completed in January 1975.

5.2 System before Modification

Figures 35 and 36 show the control-rod-latch circuitry before PM 404.

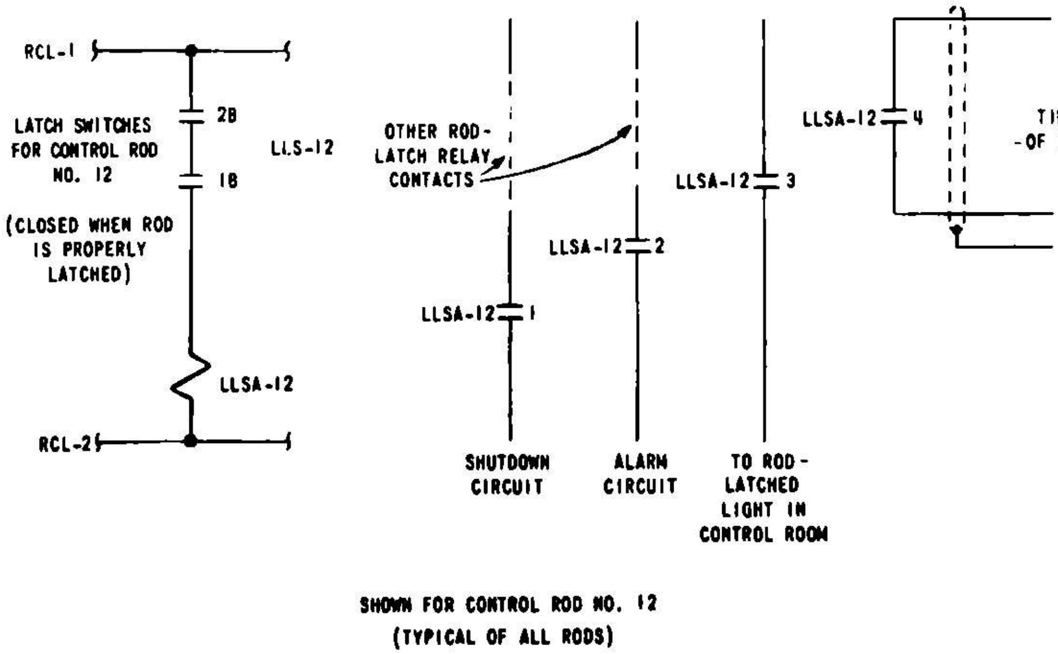


Fig. 35. Control-rod-unlatched Circuitry before PM 404

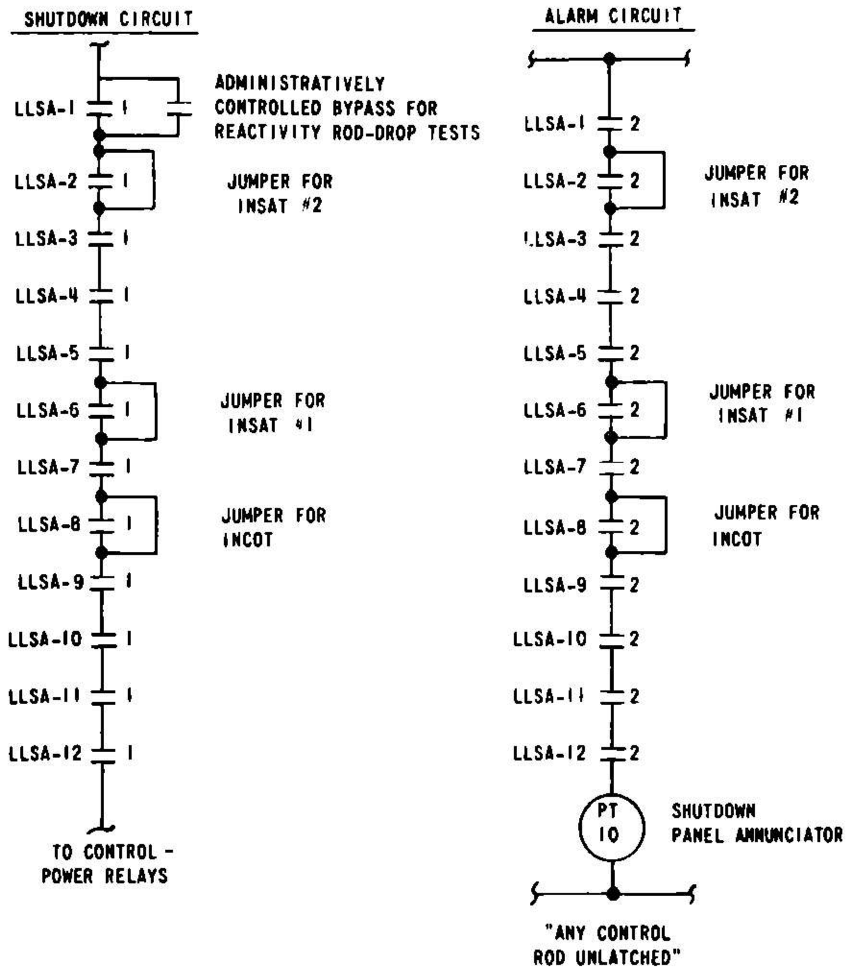


Fig. 36. Arrangement of Rod-latch Relay Contacts in the Shutdown and Alarm Circuits before PM 404

Figure 35 shows circuitry for control rod No. 12, which is typical for each control rod. Two latch switches (LLS-12-1B and -2B) are closed when the rod is properly latched and relay LLSA-12 is energized. Contacts LLSA-12-1, -2, -3, and -4 are then closed; the closing completes the shutdown and alarm circuits, the rod-latched light circuit, and the time-sequence-of-events circuit to the DAS. If either latch switch opens, relay LLSA-12 deenergizes and its contacts open, initiating a trip, actuating an ANY CONTROL ROD UNLATCHED alarm, extinguishing the rod-latch light, and transmitting a rod-unlatched signal to the DAS.

Figure 36 shows the arrangement of the rod-latch relay contacts in the shutdown circuit and the ANY CONTROL ROD UNLATCHED alarm circuit. There are jumpers around the contacts where control rods have been removed, and INCOT and INSAT experiments are inserted in their place.

5.3 System after Modification

Figure 37 shows the circuitry of the modified system, in which the control-power (CP) relay contact is in parallel with the rod-latch relay contacts in a permissive circuit. The all-rods-latched condition is a "permissive" condition required for startup. To initially energize the CP relay, all rods must be properly latched. Once the CP relay is energized, the CP contact closes and the unlatching of a rod will not activate the shutdown system.

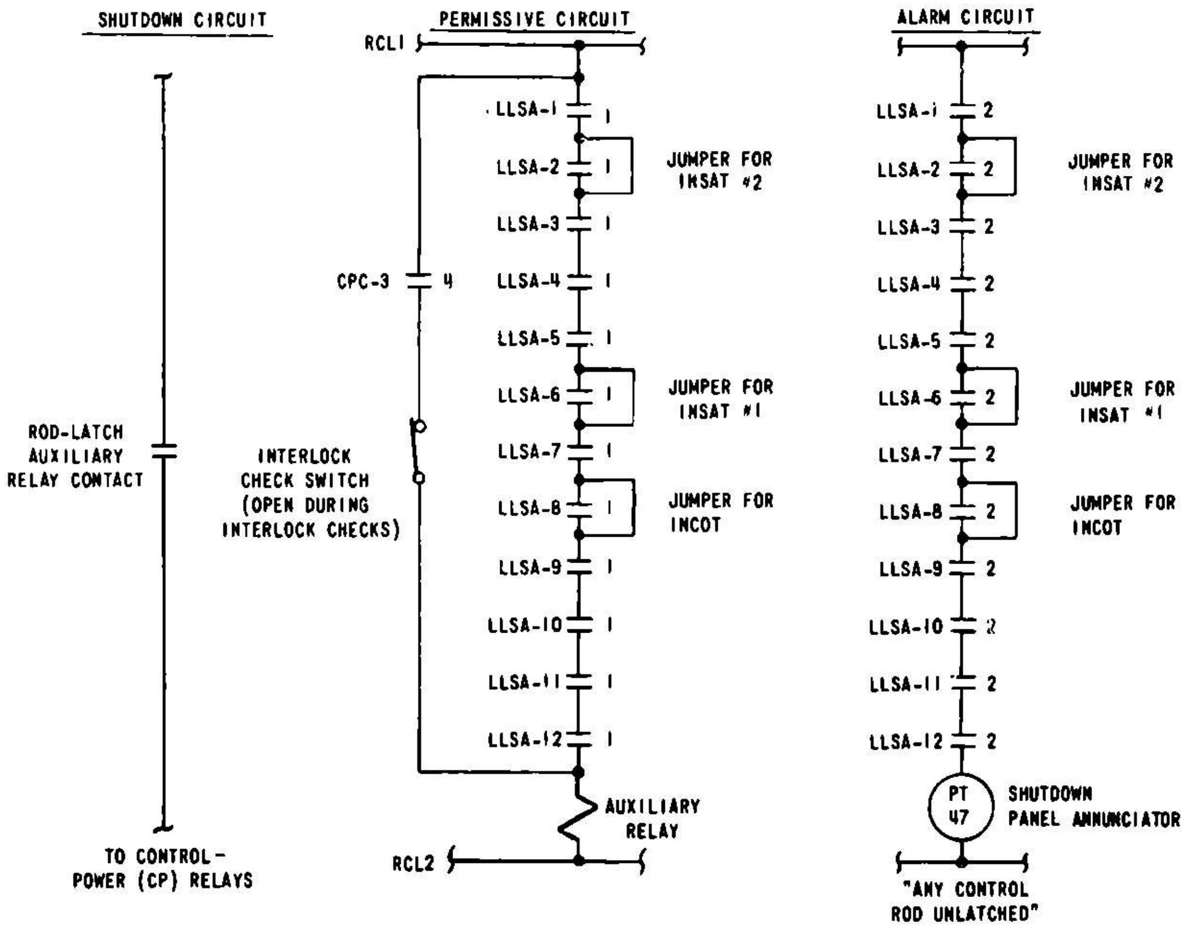


Fig. 37. Arrangement of Rod-latch Contacts in the Shutdown and Alarm Circuits after PM 404

The administratively controlled bypass contact for reactivity rod-drop tests (shown in parallel with contact LLSA-1-1 in Fig. 36) was removed. This contact was used to avoid a trip when control rod No. 1 was dropped for reactivity tests, and it is not required in the modified system.

An interlock-check-switch contact was added in series with the CP contact. This switch will be manually opened to check each latch-relay contact in the permissive circuit individually as a part of regular, quarterly interlock checks.

The alarm circuit remains the same except that a different alarm window is used to annunciate the ANY CONTROL ROD UNLATCHED condition. No changes were made to the rod-latched light circuits and the circuits to the DAS.

5.4 Justification for Modification

An automatic trip is not required when a rod becomes unlatched, and the shutdown circuit has been simplified to significantly reduce the possibility of spurious trips. As seen in Fig. 35, a failure resulting in the spurious opening of either of the rod-latch switch contacts or the latch-relay contact resulted in a spurious trip for any control rod. Nine control rods are in use; thus this modification removed 27 possible sources of spurious trips.

As concluded in Sec. 5.6, if a control rod does become unlatched, the undesirable conditions that result are safely handled by the reactor operator initiating a manual control-rod trip or anticipatory shutdown.

Because no essential safety function can be found for the trip for an unlatched control rod, removal of the trip is a step in upgrading the EBR-II shutdown circuit to bring it into compliance with RDT C16-1T[4] wherever practicable. (See Sec. 3.5, RDT C16-1T.)

5.5 Applicable Standard

RDT C16-1T[4] (specifically, paragraphs 3.5, 4.3.4, and 4.3.6) is the applicable standard for this modification.

5.6 Safety Analysis

5.6.1 Protective Function of Existing Trip

The exact protective function of the trip for an unlatched control rod is not identified in any EBR-II documentation. Presumably, this trip was intended as part of the original automatic control-rod system, which has since been removed. With such a system, if reactivity were lost because of unlatching and dropping of a control rod, the automatic rod would attempt to correct for the loss of reactivity. An automatic trip when any rod dropped would prevent this from occurring.

EBR-II does not now have an automatic power-control system, and there are no plans for such a system; thus the reactor is not susceptible to the above hypothetical incident.

It is concluded that the undesirable operating conditions that could result if an ANY CONTROL ROD UNLATCHED alarm is received can be safely handled by the reactor operator taking the appropriate action after he has confirmed that a loss of reactivity has occurred.

5.6.2 Hazards Considerations

5.6.2.1 Flux Distortion. Dropping one or more control rods will not cause core-flux distortion if a signal for an automatic trip is received. Without this trip, flux distortion will occur if the reactor remains critical. The extent of the distortion will depend on the reactivity worth of the rod that has dropped, the position of the control rods, and the specific reactor loading. However, the resulting conditions would be no worse than those that could occur if one rod were not withdrawn during startup or were inserted by a drive-system malfunction or operator error during power operation.

Calculations have shown that a fully withdrawn high-worth control rod produces variation in flux of less than 5% in adjacent subassemblies (see p. 43 of Ref. 15).

Reactor-physics data (see Table X) show that, except for control rod No. 1, when the rods are banked for normal power operation the reactor would be subcritical if a control rod were dropped, because the worth of any control rod exceeds the power-reactivity decrement. Control rod No. 1 is now used as the drop rod for routine reactivity rod-drop tests, and no hazardous flux distortions result from dropping this rod.

TABLE X. Reactor-physics Data at Beginning of Reactor Run 72A

Control-rod Positions	Control-rod Worths, Ih	Safety-rod Worth
1	4	359 Ih
2	XX07[a]	
3	224	
4	150	
5	221	
6	XX05[a]	
7	213	
8	YY02[a]	
9	228	
10	270	
11	232	
12	145	
	Total 1687	

PRD [Startup; corresponds to 11-in. (28-cm) bank position]	Excess Reactivity, Ih
92 Ih	356
Control-rod Bank Position	**Controlling Rod**
10.50 in.	12
Burnup Run 71A	
0.124 Ih/MWd	1% 441 Ih

[a]Position occupied by instrumented subassembly.

It is concluded that no hazardous flux distortion will result from dropping one or more control rods.

5.6.2.2 Binding of a Control-rod Drive. If a control-rod drive bound enough to force unlatching of the rod drive before this modification, an automatic trip would have occurred. This modification allows the reactor to continue to operate under this condition if no operator action is taken. However, the binding of a control-rod drive when the control rod is being driven up or down could not introduce an uncontrolled reactivity addition or other event that could damage the reactor core.

5.6.2.3 Secondary-pump Trip. The secondary sodium pump is tripped automatically when the reactor trips. This pump trip stops flow of secondary sodium and prevents lowering of the primary-tank temperature when a trip occurs. Dropping of a single control rod will no longer cause a trip and thus automatically affect the secondary pump. Depending on initial rod position and worth, the reactor may go to zero power in a short time. This condition would require proper response by the console operator to avoid overcooling the primary tank.

5.7 Revisions to Practices and Documentation

5.7.1 Operations

With this modification, the operator will take the following actions when an ANY CONTROL ROD UNLATCHED alarm is received:

a. Observe the reactor power level as indicated by the control-room nuclear instrumentation.

b. If reactor power is dropping, observe which control rod has dropped and start a manual control-rod trip.

c. Determine the cause of the dropped rod, and take required corrective action.

d. Follow normal procedures for recovery after a trip.

e. If reactor power is not dropping after receipt of the ANY CONTROL ROD NOT LATCHED alarm, determine the cause of the alarm and take required corrective action.

f. If the reactor is not critical, determine the cause of the alarm and take required corrective action.

This modification requires no changes to EBR-II operating limits.

5.7.2 Hazard Summary Report

No revision to Ref. 1 is required.

The following revisions to Ref. 2 are required as a result of this modification:

a. Revision to Table III (which lists system abnormalities causing trip and alarm). On page 40, item 25, remove Any Control Rod Unlatched.

b. Revision to Table K-1 (which lists system abnormalities causing alarm). On page 287, add an item to the Primary System listing to say Any Control Rod Unlatched.

c. Revision to Fig. 35 (block diagram of trip circuit). On page 139, add a dashed line around the CONTROL ROD LATCH item in the trip circuit and a reference to Note 1 for the PRIMARY COOLANT AUXILIARY PUMP and CONTROL ROD POSITION items in this figure.

5.8 Conclusion

Installation of PM 404 does not adversely affect the safety of the EBR-II reactor plant during normal operations or under any accident condition discussed in Refs. 1 and 2.

6. EBR-II PLANT MODIFICATION NO. WAF-5069: REMOVAL OF COVER-GAS TEMPERATURE AND PRESSURE TRIPS FROM REACTOR SHUTDOWN SYSTEM

E. V. Waite, N. L. Gale, and W. F. Booty

6.1 Summary

Analysis has shown that removing the trips for high pressure and high temperature of the EBR-II argon cover gas from the shutdown system does not compromise plant safety. These trips were found to be unnecessary for reactor protection and ineffective in mitigating consequences of identified incidents of concern--a sodium fire in the primary tank, and overpressurization of the primary tank due to a tube-sheet rupture in the intermediate heat exchanger (IHX).

Analysis indicates that neither pressure nor temperature trips were sensitive enough to reliably detect a sodium fire in the primary tank. Further, a fire would not endanger the reactor. The pressures resulting from a worst-case guillotine break in the IHX are unaffected by a reactor trip and are well within stress limitations of the primary tank.

This modification was completed in August 1975.

6.2 System before Modification

The premodification monitoring system for argon-cover-gas temperature and pressure consisted of a single thermal element and a single pressure transmitter, each connected to the reactor shutdown system. Either high temperature or high pressure caused reactor trip.

6.3 System after Modification

By this modification, the trips for high cover-gas temperature and pressure were removed. Figure 38 shows the affected section of the shutdown system before and after the modification, in which one jumper wire was installed and the wiring to the relays no longer required was removed.

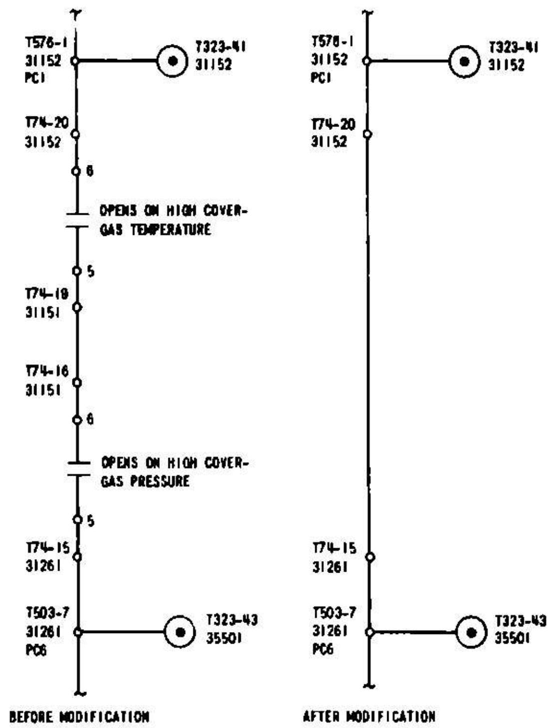


Fig. 38

Argon-cover-gas Section of Shutdown System

The annunciation of high cover-gas pressure was retained, with the same annunciator window at instrument control center No. 1 (ICC 1) being used. The annunciator windows on the trip section of the console annunciator for high cover-gas temperature and pressure were removed, and a window for each condition was provided in the alarm section of the console annunciator.

The following were unchanged: system arrangement, meters and recorders in the control room and at instrument panel No. 1, alarms other than those annunciators described above, and test sources and setup procedures for the instrumentation for cover-gas temperature and pressure.

6.4 Justification for Modification

The purpose of this modification is to upgrade the PPS by:

a. Reducing the number of trip contacts in the system.

b. Reducing the susceptibility of the plant to spurious trips without compromising plant safety.

c. Complying with Sec. 3.5 of RDT C16-1T[4] in removal of "anticipatory" reactor trips.

In this modification, the trips for cover-gas temperature and pressure were removed from the shutdown system; the safety analysis (see Sec. 6.6) justifies removal of these trips because they did not perform a necessary safety function. Removing them eliminated sources of spurious trips and has brought the EBR-II shutdown system into better conformance with Standard C16-1T.[4]

6.5 Applicable Standard

This modification is governed by the requirements of RDT C16-1T;[4] it requires no variances to these requirements.

6.6 Safety Analysis

6.6.1 Potential Events and Related Hazards

The events of concern for temperature and pressure of the argon cover gas are (a) a sodium fire in the primary tank and (b) overpressurization of the tank. Safety limits established to ensure primary-tank integrity require that the differential pressure of the cover gas (with respect to building static pressure) be ≤ 250 in. H_2O ($\sim 6.2 \times 10^4$ Pa) (based on design criteria for the pressure-relief system).

Analysis confirmed that the trips on cover-gas pressure and temperature are not needed either to mitigate the consequences of the incident or to protect the reactor from postulated damage. The only credible overpressurization incident identified (see Sec. 6.6.3.3) is the tube-sheet rupture of inlet piping of the intermediate heat exchanger. The argon-gas high-pressure sensor would have detected this incident, but reactor trip would have no mitigating effect because consequences of the incident are independent of reactor operation. A detailed analysis of this incident appears in Appendix B.

The instrument channels could not have detected a fire in the primary tank unless the fire had been of extraordinary size. The Design Basis Accident (DBA), in which the primary tank is assumed to fail,[1] is the only mechanism identified as a source of such a large fire. The pressure monitor could not detect a fire in the primary tank, but the pressure monitor might have detected the shock wave associated with the DBA. This is so because the control system for cover-gas pressure can be expected to nullify any pressure changes resulting from a fire with the primary system intact. The temperature monitor could detect only a very large fire in the primary tank. The inability of these channels to detect a fire with the primary tank intact is discussed in Appendix C. The DBA would destroy the reactor core and prevent trip action before either the temperature or pressure could change.

In addition to showing that the subject instrument channels are not likely to detect fires in the primary tank, Appendix C shows that a trip would not influence a fire in the primary tank nor would the fire present an immediate hazard to the reactor. Analysis indicates that the rate of Na_2O buildup at the maximum combustion rate is less than 0.03 in./h (2.1×10^{-7} m/s). Such a buildup would not prevent the control and safety rods from moving before the fire could be detected. Appendix C also shows that the temperature gradient expected during a sodium fire would be so large (~ 500°F/in. or 11.3°C/mm) that the primary-tank cover would be heated very little and binding of the control rods could not be caused by excessive temperature differential in the cover.

The protection channels for primary-cover-gas pressure and temperature were not in compliance with Sec. 3.5 of RDT Standard C16-1T,[4] because these monitors were anticipatory and only warned of a possible impending accident.

6.6.2 Historical Background

The instrument channel that caused trips because of primary-cover-gas pressure had caused spurious trips five times during critical operation and ten times during noncritical operation. The primary-cover-gas-temperature channel had caused one spurious trip.

Protective action to prevent or mitigate the consequences of a sodium fire or an overpressurization in the primary tank has not been required, because the events have not occurred. [Overpressurization is defined as a pressure greater than 14.5 in. water ($\sim 3.5 \times 10^3$ Pa) above the reactor-building static pressure.] That they did not occur is very significant, because it indicates that the primary-cover-gas system and its interface systems are adequately designed to prevent such occurrences.

All EBR-II plant incidents since March 1969 have been recorded. Only one case (Incident No. 63, February 21, 1971) of air injection into the primary cover gas has occurred. This incident was caused by a leak in loop B of the fission-gas-monitor (FGM) system. The event was discovered when loop B was found to be plugged. The leak was confirmed by indication of high nitrogen concentration on the primary-gas chromatograph. This incident was of minor consequence, because the total air volume injected was very small. Total FGM flow is about 5 cfm ($\sim 2.4 \times 10^{-4}$ m^3/s), and the leak in loop B was considerably smaller than the total loop flow. The instrumentation for primary-cover-gas pressure and temperature did not detect this leakage of air into the system.

To ensure that the recurrence of a similar event would be detected more rapidly, a primary-gas-chromatograph alarm was installed in the control room.

From the available data, the probability of a major incident involving a cover-gas fire has been postulated as less than 11×10^{-6} per hour. This value is a conservative upper limit or boundary that is used in lieu of actual failure data. The number is obtained by (a) observing 10 years of operation without a cover-gas fire and (b) assuming the event could occur tomorrow.

The probability of a system failure causing a primary-tank overpressurization from causes other than a cover-gas fire are established in the same manner. That is, no single recorded incident indicates a serious overpressurization. The incidents recorded involve only small transient-pressure excursions caused by manipulations of the FUM argon-system valves.

6.6.3 Analytical Investigation

Figures 39 and 40 are fault trees showing how high temperature or overpressurization could occur in the cover-gas system. Figure 39 identifies possible events that could lead to a fire in the primary tank. Figure 40 identifies those that could lead to a cover-gas overpressurization.

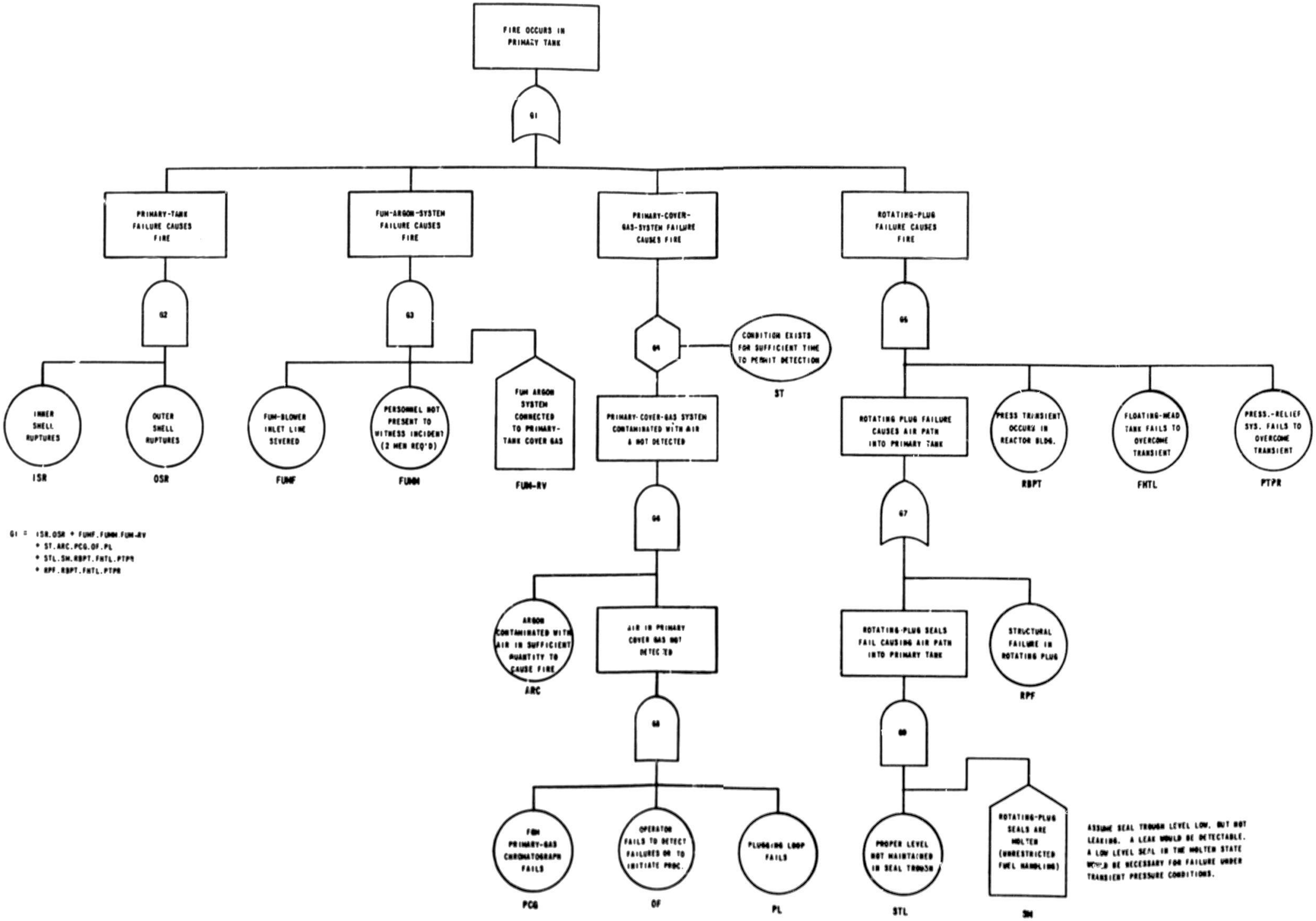
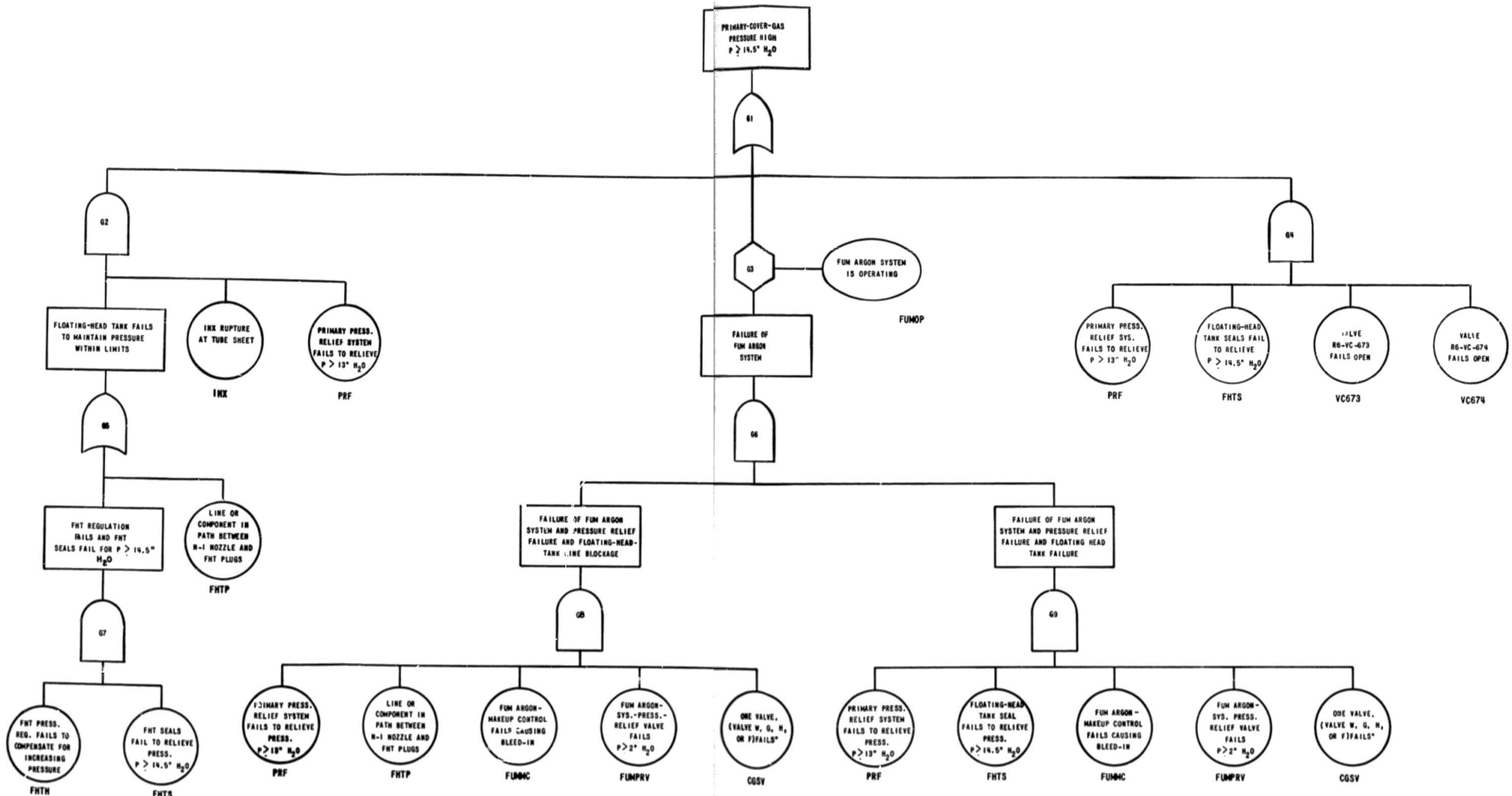


Fig. 39. Fault Tree of Events Causing Primary-tank Fire



G1 = PRF.FHTP.IHX + PRF.FHTS.FHTH.IHX
+ PRF.FHTS.VC673.VC674 +
+ PRF.FHTS.FUMOP.CGSV.FUMMC.FUMPRV + PRF.FHTP.FUMOP.FUMMC.FUMPRV.CGSV

* IN 86% OF ALL FUM ARGON SYSTEM VALVE LINEUPS, A SINGLE SPECIFIC VALVE POSITION CHANGE WOULD CAUSE THE OUTLET LINES TO BE CLOSED WHILE AN INLET LINE WAS OPEN.

Fig. 40. Fault Tree of Events Causing High Pressure in Primary-tank Cover Gas. Conversion factor: 1 in. H_2O = ~24.8 Pa.

Table XI contains identification of component code names and the failure history for important components.

TABLE XI. Data Listing for Logic Diagram

Code Name	Fault	Failure History
ISR	Inner shell of primary tank ruptures.	None
OSR	Outer shell of primary tank ruptures.	None
FHTL	Floating-head tank fails, causing low pressure in primary tank.	One recorded; nonscheduled maintenance call.
RPF	Rotating plug ruptures owing to structural failure.	None
FUMM	Operating personnel are not present during fuel handling to witness incident.	None. Operating procedures require two operators present in the reactor building and on the operator floor during fuel-handling operation.
FUMF	Inlet line of FUM blower sustains double-ended offset shear while in use (FUM connected to fuel-transfer port).	None
FUM-RV	FUM argon system is connected to the reactor-vessel cover gas.	This is not a failure, but a condition necessary to enable an accident sequence. This condition is estimated to occur 30% of the time.
SM	Rotating-plug seal is molten.	This is not a failure, but a condition necessary to enable an accident sequence. Assume a rate of four days per month.
STL	Proper level is not maintained in seal trough.	None
VC673	Valve R6-VC-673 fails open.	None
VC674	Valve R6-VC-674 fails open.	None
PTRR	Primary-tank pressure regulator fails to relieve reactor-building pressure transient in primary tank.	None
IHX	Intermediate heat exchanger ruptures at tube sheet.	None
PRF	Primary pressure-relief system fails to relieve pressure >13.0 in H_2O (~3.2×10^3 Pa).	None
FHTP	Line or component in path between N-1 nozzle and floating-head tank plugs.	One recorded; nonscheduled maintenance call.
FHTS	Floating-head-tank oil seals fail to relieve pressure >14.5 in H_2O (~3.6×10^3 Pa).	None
FUMMC	FUM argon makeup control fails, causing argon bleed-in.	None
FUMPRV	Pressure-relief valve of FUM argon system fails.	None
CGSV	One remote-control valve fails in a specific valve position to cause FUM argon system to be in lineup for possible incident.	Six valves control valving lineup. Eighty-six percent of lineup configurations could revert to failed condition by one valve failing.
FUMOP	FUM argon system is operating.	This condition is not a failure but is necessary for an accident. System usually remains on unless it is not needed for more than three days.
RBPT	Pressure transient occurs in reactor building.	None
ARC	Argon system is contaminated with enough air to cause a fire.	None

TABLE XI (Contd.)

Code Name	Fault	Failure History
PCG	FGM primary-gas chromatograph fails.	Eleven recorded; nonscheduled maintenance calls.
OF	Operator fails to detect failure in chromatograph or plugging loop, or operator fails to initiate proper procedure.	None
PL	Plugging loop fails.	One recorded; nonscheduled maintenance call.
ST	Failure in cover-gas system causing fire exists for sufficient time to permit fire detection.	A necessary condition for a fire resulting from a failure in the primary-cover-gas system.
FHTH	Pressure regulation for floating-head tank fails to compensate for increasing pressure.	No recorded incident; however, a large volume associated with a pressure increase such as that associated with a complete severence of the IHX tube sheet could result in a failure of the floating-head tank to regulate pressure.

6.6.3.1 Analytical Boundaries. This analysis, because it is limited to cover-gas temperature and pressure, assumes the following boundaries: (a) No response of other protective functions is assumed to terminate the condition; (b) the only events considered are those that, it they occurred, could compomise the integrity of the primary-cover-gas system. In addition to the above assumptions, loss of instrument air is not considered an unsafe failure. This boundary is factually established because all the pneumatic valves included in this analysis fail in the safe position upon loss of instrument air.

Because the cover-gas parameters are common to both the fuel-handling and operating shutdown systems, this analysis makes no distinction between the two operating modes; the incidents assumed could happen in either state of operation

6.6.3.2 Analysis of Cover-gas Fire. The cut sets[16] obtained from Fig. 40 for the fault tree for cover-gas fires are:

$$\begin{aligned} \mathrm{FIRE} = {} & \mathrm{ISR \cdot OSR + FUMM \cdot FUMF \cdot FUM\text{-}RV} \\ & + \mathrm{ST \cdot ARC \cdot PCG \cdot OF \cdot PL} \\ & + \mathrm{RPF \cdot PTPR \cdot FHTL \cdot RBPT} \\ & + \mathrm{STL \cdot SM \cdot PTPR \cdot FHTL \cdot RBPT}. \end{aligned}$$

No single-event cut set appears in this reduction. One second-order cut set appears in the equation. This cut set (ISR·OSR) is the postulated rupture of both the inner and outer shells of the primary tank. Only the DBA could conceivably cause this extremely unlikely incident, in which the resulting core meltdown would obviously make a reactor trip of no consequence.

One third-order cut set appears in the fault tree for a cover-gas fire. This event involves the FUM system initiating the accident (blowing air into the primary tank) by complete severance of the FUM argon blower inlet line. This cut set is (FUMM·FUMF·FUM - RV).

The event described by this cut set requires all operating personnel to be absent from the reactor building during fuel handling; the FUM argon system must be connected to the primary cover gas, and the FUM-blower inlet line must be completely severed.

The remainder of the fault tree for a cover-gas fire is composed of one fifth-order cut set in which the cover-gas makeup system initiates the fire, and one fourth- and one fifth-order cut set in which a cover-gas fire results from rotating-plug failures. These cut sets are of such a high order that they will not be considered.

The only events of interest are (a) the primary-tank rupture, which, as previously stated, could only be caused by a DBA; and (b) the severance of the FUM-blower inlet line, a third-order event, which, as discussed in Appendix C, cannot be reliably detected by the instruments for cover-gas temperature and pressure.

6.6.3.3 Analysis of Primary-tank Overpressurization. The cut sets obtained from Fig. 41 for the fault tree for high pressure in the primary tank are:

$$\text{TOP} = \text{PRF}\cdot\text{FHTP}\cdot\text{IHX} + \text{PRF}\cdot\text{FHTH}\cdot\text{FHTS}\cdot\text{IHX}$$

$$+ \text{PRF}\cdot\text{FHTS}\cdot\text{VC673}\cdot\text{VC674}$$

$$+ \text{PRF}\cdot\text{FHTS}\cdot\text{FUMMC}\cdot\text{FUMPRV}\cdot\text{CGSV}\cdot\text{FUMOP}$$

$$+ \text{PRF}\cdot\text{FHTP}\cdot\text{FUMMC}\cdot\text{FUMPRV}\cdot\text{CGSV}\cdot\text{FUMOP}$$

The equation contains one third-order, two fourth-order, and two sixth-order-cut sets. The single third-order cut set considers overpressure in the primary tank resulting from a rupture in the intermediate heat exchanger (IHX). This cut set (PRF·FHTP·IHX) indicates that, for overpressurization to occur as a result of an IHX rupture, both the pressure-relief system and the floating-head tank would have to fail to control the pressure excursion. In this case, the floating-head tank is assumed to fail as a result of plugging of the connecting line (or its components) between the primary tank and the floating-head tank. However, if we consider the complete rupture of the IHX tube sheet as analyzed in Appendix B, the floating-head tank cannot be claimed for protection, and the cut set degenerates to a second-order event (PRF·IHX). In this case, protection against overpressure is solely by the pressure-relief system, because the floating-head tank is not designed to accommodate the large flow rates resulting from a complete rupture of the IHX tube sheet.

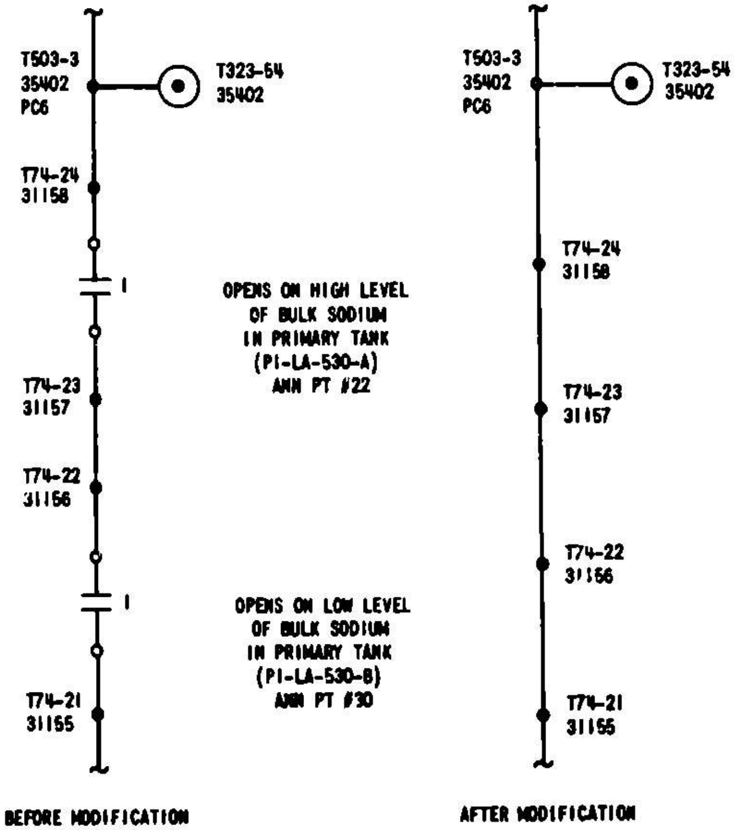


Fig. 41. Bulk-sodium-level Portion of Shutdown System

One fourth-order cut set also involves the IHX rupture, in which the pressure-relief system, the pressure regulation for the floating-head tank, and the seals of the tank all would fail to relieve the pressure transient. If we again assume a complete rupture of the IHX tube sheet, the floating-head tank cannot be claimed as protection. In this instance, the fourth-order cut set degenerates to second order (PRF·IHX). Note that this cut set is identical with the previous second-order (degenerated) set.

The remaining fourth-order cut set involves failure of the argon supply system, in which case, if the bleed-in valves (VC-673 and VC-674) were to fail open, the pressure-regulating capacity of the floating-head tank would be lost. The further failures of the pressure-relief system and the floating-head-tank seals to relieve (in itself unlikely) would result in overpressure. The failure of an oil seal to relieve itself is much more remote than operating experience indicates. The more obvious failure of this type of a seal would be that of a low oil level causing a pressure leak.

The sixth-order cut sets are for cover-gas overpressure resulting from failures in the argon makeup system of the FUM. Because of the high order of these sets, these faults are extremely unlikely.

Thus, the fault tree for cover-gas overpressurization indicates only one incident of concern: rupture of the IHX tube sheet. This incident is analyzed in Appendix B, where the results of a complete tube-sheet severance are presented. The results in Appendix B indicate that secondary sodium would drain into the primary tank for about 48 s. During this interval, a maximum of 3500 gal (~13.2 m^3) of secondary sodium would flow into the primary tank and the tank level would rise 11 in. (~29 cm). [The bulk sodium level in the primary tank is normally 15 in. (~38 cm) below the top cover.] Cover-gas pressure would peak (depending on the actual friction factor and operability of the pressure-relief system) at a differential pressure of 163-265 in. of water (41-66 kPa) about 30 s into the transient. For the maximum pressure, no pressure relief is assumed, i.e., complete failure of the pressure-relief system. Thus, the fact that this pressure exceeds the limit established to protect this system is of no concern here.

This analysis considers the extreme hypothetical case of the IHX severing completely at the tube sheet. The assumptions used in Appendix B were made to permit analyses in a reasonable time interval. Complete severence without flow resistance is extremely unlikely, and even a badly damaged IHX would have appreciable flow impedance. This resistance would both damp and reduce the transient effects of the incident.

The remaining event of concern in this analysis is the peak pressure, if we consider the increase in tank head due to an additional 11 in. of sodium (~2.3 kPa) and add this to the peak gas-pressure values--163 and 265 in. of water (~41 and ~66 kPa). These values can be used in accordance with the results presented in Appendix D* to calculate tensile stress on the inner vessel bottom.

a. At 9.6 + 0.34 = 9.94 psig (~68.5 kPa gauge), the tensile stress is 19,123 psi (~132 MPa).

b. At 5.9 + 0.34 = 6.24 psig (~43 kPa gauge), the tensile stress is 16,280 psi (~112 MPa).

The allowable working stress intensity permitted by the ASME Code at 750°F (~390°C) is 14,700 psi (~101 MPa). This value has a safety factor of 5/8 of minimum yield at temperature.[17] This indicates a yield strength greater than 23,500 psi (~162 MPa). Although the analytical stress levels indicate that the stress intensity may approach the ASME working limit for a short interval, the calculations indicate that the primary tank would not be deformed or damaged and its service life would not be reduced.

The tank level increasing 11 in. (~28 cm) will not cause the sodium to contact the top cover. Thus the control rods will not be subject to any binding due to thermal stress that could prevent them from moving. The duration of the pressure transient is of no major concern for the level change.

6.7 Revisions to Practices and Documentation

6.7.1 Operations

EBR-II operating limits were revised to delete pressure and temperature of argom cover gas as parameters for reactor trips.

---

*The relationship is $\frac{(13 + P)\ \text{psig}}{13\ \text{psig}\ (\sim 90\ \text{kPa})}$ [11,000 psi (~75,845 kPa)] = tensile stress (psi).

Operating instructions were changed to reflect the removal of the trips and the trip alarms. Emergency procedures were changed to provide operational intervention if either high temperature or pressure occurs in the primary cover gas. The prescribed actions include validation of the alarm, initiation of an anticipatory shutdown within the limits defined in the EBR-II technical specifications, and elimination of the source of pressure or fire if possible.

6.7.2 Administrative Control

Information on cover-gas temperature and pressure was unchanged. Because these parameters for plant protection are anticipatory, they now serve a more useful role--informing operators of the possibility of hazardous conditions developing.

6.7.3 Technical Specifications

Technical specifications did not require revision.

6.7.4 Hazard Summary Report

The following sections of Refs. 1 and 2 are superseded:

ANL-5719 (Ref. 1)	ANL-5719 Addendum[2]
Sec. IV.A.3.b(2)	Table III
Table VII	Fig. 35

6.8 Conclusion

The trips for high temperature and high pressure of primary cover gas were inadequate and, with respect to RDT Standard C16-1T, unnecessary. These scrams apparently originated from the philosophy used during the conceptual reactor design period. All the reactor performance and response characteristics were not firmly established, and any variable that could initiate a trip during any postulated abnormal condition was included in the shutdown system. The primary mission of EBR-II has shifted from a reactor experiment to irradiation facility, and the reactor performance and response characteristics are now better understood. Therefore to increase plant availability trip channels demonstrated inadequate and unnecessary have been eliminated to increase plant availability.

The basic safety of the reactor with respect to cover-gas overpressurization has been improved since the original design, most significantly by adding a pressure-vacuum relief system to the primary tank. The system protects against pressure transients resulting from the high-volume flow rate that would accompany a hypothetical large IHX rupture.

A cover-gas fire could not be reliably detected by the instrumentation for cover-gas temperature and pressure instrumentation. More importantly, a cover-gas fire would not endanger the reactor or inhibit the reactor control-rod system from completing an orderly reactor shutdown. Overpressurization, due to a large IHX rupture, would not endanger the reactor as a result of a change in bulk sodium level or additional primary-vessel stress. A cover-gas fire would be detected through the primary-gas-chromatograph alarm on high nitrogen content. In the worst-case fire, the chromatograph would alarm the condition within 4 min. The primary-sodium plugging temperature and the continuous readout from the oxygen-hydrogen meter module would be useful in detecting smaller air leaks.

Thus, this modification has not adversely affected the safety of the EBR-II reactor plant during normal operations or under any accident conditions discussed in Ref. 1 or previous addenda.

7. EBR-II PLANT MODIFICATION WAF-5087: REMOVAL OF BULK-SODIUM TEMPERATURE AND LEVEL TRIPS FROM REACTOR SHUTDOWN SYSTEM

W. F. Booty, L. K. Chang, H. M. Forehand, Jr.,*
W. K. Lehto, and E. V. Waite

7.1 Summary

Analysis has shown that removing the trips for high and low levels and high temperature of primary bulk sodium does not compromise plant safety. These trips were found unnecessary for reactor protection and ineffective in mitigating consequences of identified incidents of concern: tube-sheet rupture in the IHX, primary-tank failure, and loss of secondary-system cooling.

Analysis shows that the rise in bulk-sodium level due to a postulated guillotine break in the IHX would not present a reactor-safety concern; primary sodium would not contact the reactor cover, and resultant stresses are well within allowable limits for the primary tank. The only identified mechanism that could conceivably cause the bulk-sodium level to drop below primary-pump inlets is the DBA, for which a trip on bulk-sodium low level is ineffective. The temperature rise due to a loss of secondary cooling would not cause either fuel or coolant overtemperatures or unacceptable primary-tank or IHX stresses.

This modification was completed in September 1975.

7.2 System before Modification

The premodification monitoring system for bulk-sodium level and temperature consisted of one level sensor and four thermocouples, each connected to the reactor shutdown system. Either high or low level, or high temperature sensed by two out of four thermocouples, caused reactor trip.

7.3 System after Modification

In this modification, trips for high and low levels and high temperature of primary bulk sodium were removed. Figure 42 shows the affected section of the shutdown system before and after the modification.

The following were unchanged: system arrangement, meters and recorders for sodium level and temperature, test sources, and setup procedures for the instrumentation. The trip annunciators were moved to the alarm section of the reactor-console annunciator.

*Los Alamos Scientific Laboratory.

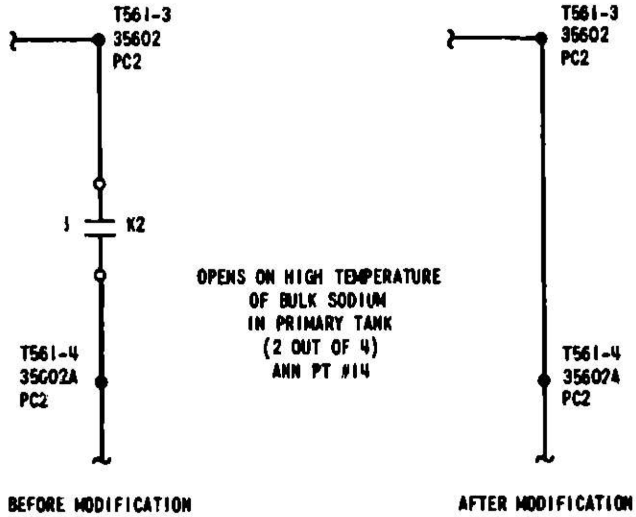


Fig. 42

Bulk-sodium-temperature Portion of Shutdown Circuitry

7.4 Justification for Modification

The purpose of this plant modification is to upgrade the PPS by:

a. Reducing the number of trip contacts in the system.

b. Reducing the susceptibility of the plant to spurious trips without compromising plant safety.

c. Complying with Sec. 3.5 of RDT C16-1T[4] in removal of "anticipatory" trip variables.

Reducing the number of spurious-trip sources through the removal of the trips for high and low levels of bulk sodium and the high-temperature trip enhances plant availability without compromising plant safety.

7.5 Applicable Standard

This modification is governed by the requirements of RDT C16-1T;[4] it requires no variances to those requirements.

7.6 Safety Analysis

Before this modification, trips associated with parameters for primary bulk sodium were:

a. High Level. A trip was initiated at the 302-in. (7.7-m) level. The condition was sensed by level transducer R1-LT-530R.

b. Low Level. A trip was initiated at the 294-in. (7.5-m) level. The condition was also sensed by level transducer R1-LT-530R.

c. High Temperature. Contact K2-1 opened on two-out-of-four indications of bulk-sodium temperatures higher than 710°F (377°C). The two-out-of-four logic received input from the following instrument channels: R1-TC-540AR, -540AS, -540AV, and -540AT.

Normal bulk-sodium level is 299 in. (7.59 m). The +3- to -5-in. (+7.6- to -12.7-cm) differential between the normal level and the high- and low-level trips includes compensation for thermal expansion and contraction. Normal bulk-sodium temperature is 700°F (371°C). The trip for high temperatures was set to actuate at 10°F (-12°C) above normal.

7.6.1 Historical Background

Six reactor trips have occurred since 1958 as a result of bulk-sodium level and temperature monitors. Out of this total, four resulted from spurious high-temperature trips, the remainder from spurious low-level trips. The high-level trip has never been actuated. The bulk-sodium conditions for which the trips were provided have not occurred during the plant lifetime.

7.6.2 Analytical Investigation

The hypothetical plant incidents that could lead to high or low levels of sodium are developed in the two fault trees of Fig. 43. The fault tree in Fig. 44 depicts the incidents that could cause a high bulk-sodium temperature. The following limits are listed to establish definitive analytical boundaries:

High level:	314 in. (7.98 m)
Low level:	222 in. (5.64 m)
High temperature:	800°F (427°C)

The listed values for high level are not related to operating limits, but represent absolute limits that, if exceeded, could cause plant damage. A 15-in. (38-cm) level increase--from 299 in. (7.59 m)--would cause bulk-sodium contact with the primary-vessel top cover. No safety problems would occur for bulk-sodium levels above normal until contact with the top cover is established. This contact would cause excessive temperature increases in the top cover, resulting in bowing of the cover and possible control-rod binding. At the low-level limit--222 in. (5.64 m)--the primary-pump inlets would become uncovered, primary-coolant flow would be lost, and serious overheating of the core would result under conditions of either reactor operation or shutdown.

The high-temperature limit of 800°F (427°C) for bulk sodium represents a safe temperature and has been selected in the absence of data to support a higher limit. At this temperature, the primary vessel is not thermally overstressed. That this temperature is a "known-safe" limit is substantiated in Appendix E.

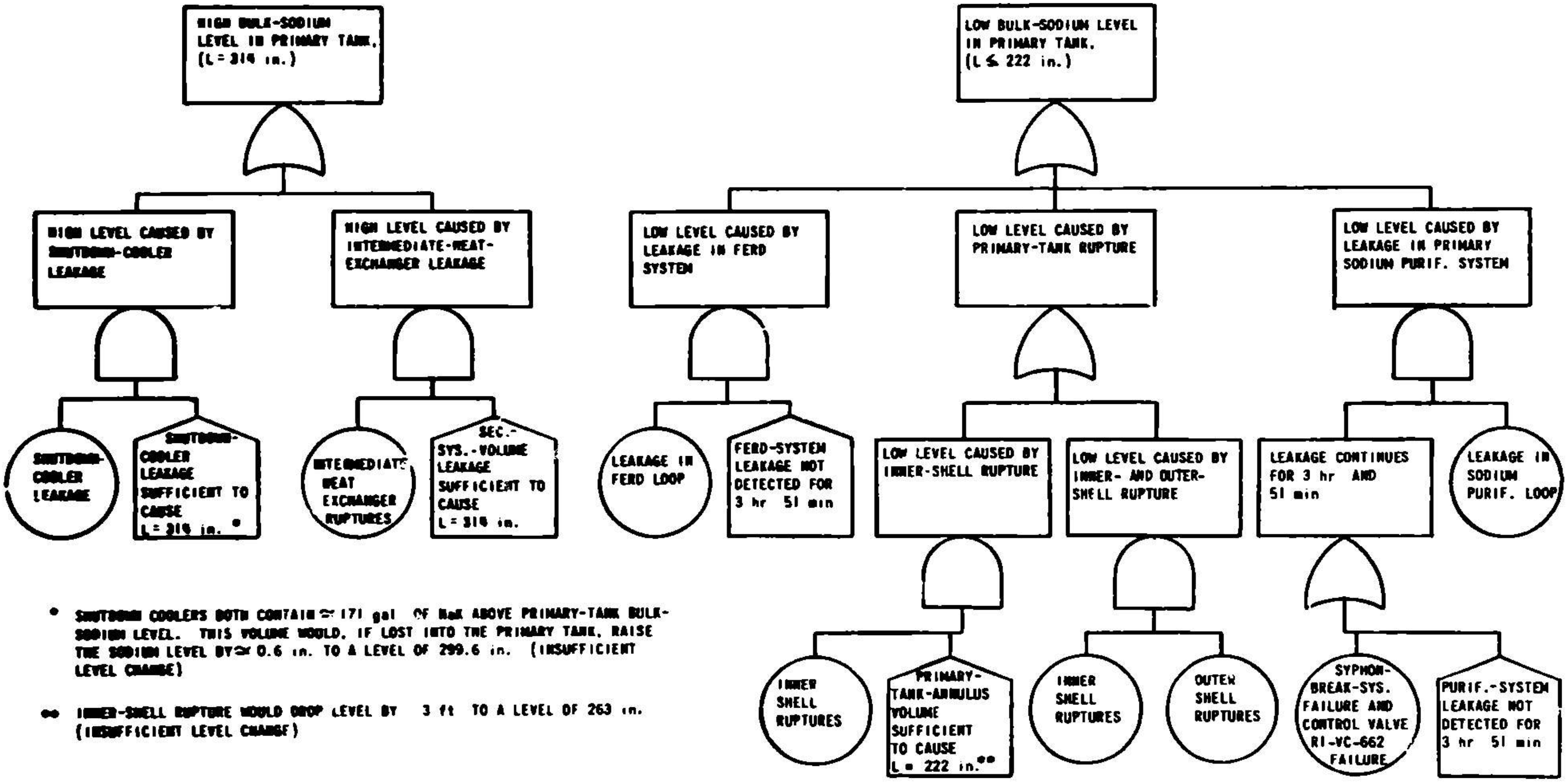


Fig. 43. Fault Tree of Events Leading to High and Low Levels in Primary Bulk Sodium. Conversion factors: 1 in. = 25.4 mm; 1 gal = 3.785 x 10^{-3} m^3; 1 ft = 0.3048 m.

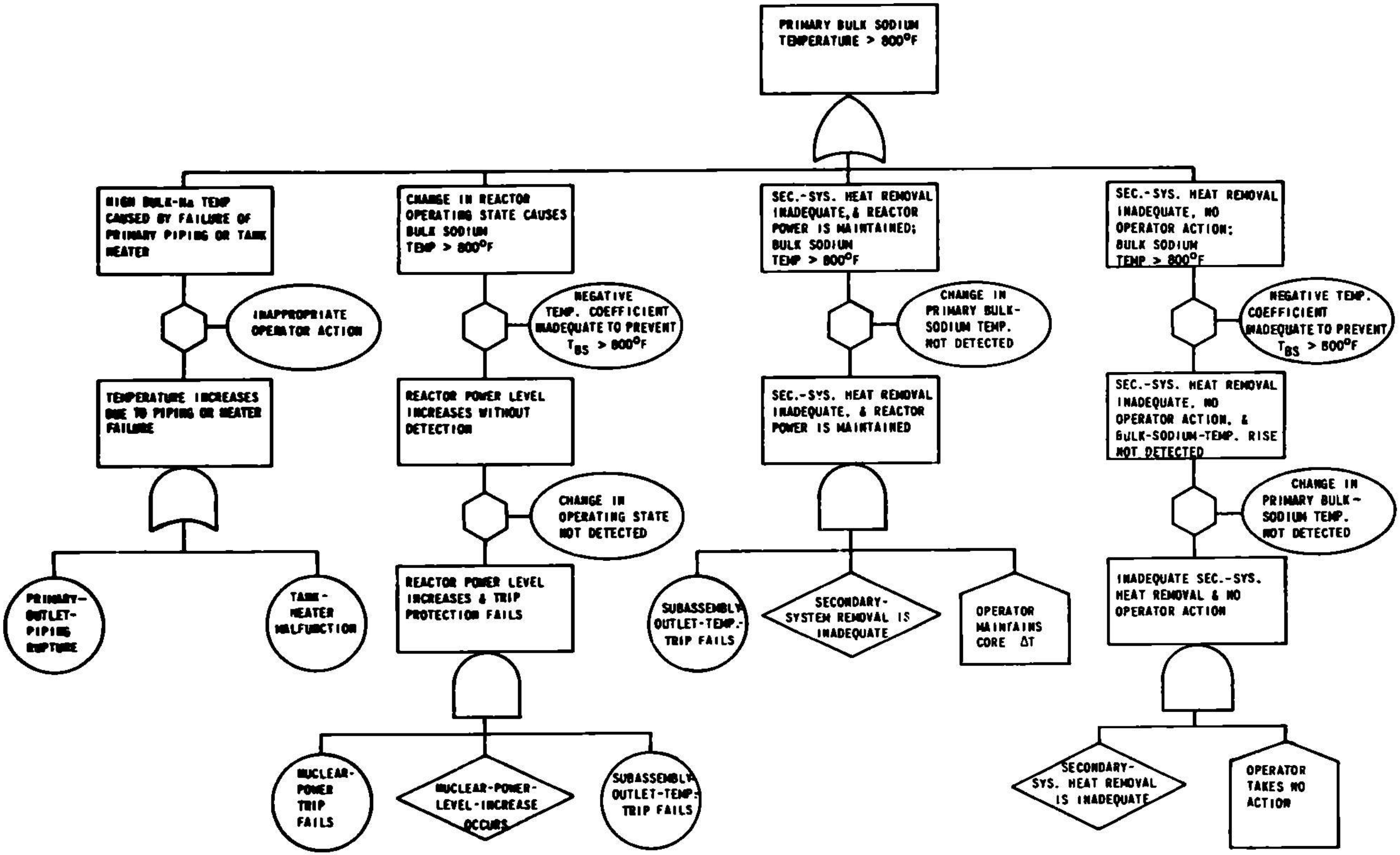


Fig. 44. Fault Tree of Events Leading to High Temperature in Primary Bulk Sodium. Conversion factor: °C = (°F − 32)/1.8.

7.6.2.1 High Level. The fault tree for high bulk-sodium (Fig. 43) indicates that the only identifiable way that a high level could occur is by a rupture of the IHX. The shutdown coolers were investigated to determine whether a shutdown-cooler leak into the primary tank could cause a significant level change in the primary bulk sodium. The results indicated that both coolers contained (above the primary-tank sodium level) about 171 gal (0.65 m^3) of NaK. Adding this volume of NaK to the primary tank would raise the level of the primary bulk sodium about 0.6 in. (1.5 cm). It is concluded that a leak into the primary tank from either or both shutdown coolers would not significantly increase the bulk-sodium level in the primary tank.

7.6.2.2 Low Level. Figure 43 indicates three possible causes for a low level of bulk sodium: (1) a rupture in the primary tank, (2) a leak in the sodium-purification system, and (3) a leak in the FERD loop.

A rupture of the inner shell of the primary tank could not decrease the bulk-sodium level more than 3 ft (0.91 m), which is not enough to uncover the primary-pump inlets. A rupture of both inner and outer shells could cause the level to drop below the primary-pump inlets; the resulting loss of primary suction would initiate a trip. Failure of both shells could cause primary sodium to become exposed to air in the shield cooling system.

A low level could be caused by leakage from the primary-sodium purification system. The suction-line inlet of this system is about 15 ft (4.6 m) below the normal bulk-sodium level; the primary-pump inlets would be uncovered at the 222-in. (5.64-m) level--a drop of 6.4 ft (1.95 m). The purification-system pump is rated at 96 gpm (0.006 m^3/s). At this flow rate, the primary-pump inlets would be uncovered if the leak rate is assumed equal to the rated capacity, after about 3 h, 51 min.

Because a long interval is required for a significant drop in level and equipment is available to detect change in level, reactor-building radiation activity, and reactor-building fire, the protective margin is adequate to permit operator response to a purification-system leak. Because part of the purification system is below the bulk-sodium level, a leak in the system could cause primary sodium to siphon from the loop after pump shutdown. A siphon-break system has been installed to eliminate the problem of leakage after pump shutdown.

Failure of the fuel-element-rupture-detector (FERD) loop can also cause leakage of sodium from the primary tank. FERD-loop siphoning is not possible because (a) the FERD is above the bulk-sodium level, and (b) a rupture in the inlet side the loop would not cause leakage, since the loop is under negative pressure. However, a rupture in the discharge side of the loop could cause leakage. Because the FERD takes suction at the outlet of the IHX, a rupture on the discharge side of the FERD loop could hypothetically lower the bulk-sodium level by about 9.6 ft (2.9 m). The primary-pump inlets would be uncovered if the level dropped 6.4 ft (1.9 m). At the nominal FERD pumping rate of 100 gpm (0.0063 m^3/s), about 4 h would be required to uncover the primary-pump inlets. As with the primary-sodium purification loop, time and alarms are adequate for operator response.

7.6.2.3 High Temperature. A malfunction of the secondary system or a change in the reactor operating state can increase bulk-sodium temperature.

High bulk-sodium temperature--T ≥ 800°F (427°C)--due to an undetected operating-state change is believed extremely unlikely because of the trips and alarms available to detect a change in operating conditions.

Figure 44 shows the possibility of high temperature of bulk sodium due to secondary-system failure, e.g., if the reactor thermal power were transferred directly to the primary tank. This incident could happen as a result of rupture or blockage of the secondary system, a failure of a secondary-system pump, or spurious drainage of the secondary-system sodium into the secondary-sodium storage tank. For these events to occur undetected is considered extremely unlikely.

The fault tree also includes lesser events, such as rupture of primary-system outlet piping and malfunction of tank heaters, which, if allowed to proceed undetected, might result in high bulk-sodium temperature.

Other events in the tree are more limiting; analysis of them provides the basis for safety. Rupture of primary-system piping is less severe because IHX integrity is maintained and partial bulk-sodium cooling would occur either by forced convection or, in the extreme case, by natural convection. The partial cooling would mitigate the transient and allow more time for operator action.

With malfunction of a primary-tank heater, the maximum temperature rise is only 2-3°F/h (1-1.7°C/h), whereas, the analysis reported later considers a 0.6°F/s (0.33°C/s) transient, which is clearly more severe.

7.6.3 Analytical Results

The fault trees in Figs. 43 and 44, combined with the discussion of Sec. 7.6.2, indicate three possibilities of concern for abnormal bulk-sodium level and temperature:

a. An IHX rupture overpressuring and overstressing the primary vessel. An additional concern is that the increasing bulk-sodium level, due to secondary-sodium inleakage, might cause the bulk sodium to contact the primary-tank top cover. This would cause thermal differentials that would in turn generate stresses in the top cover and the rotating plugs. These stresses might cause sufficient deformation in the rotating plugs to prevent trip action of control rods.

b. A primary tank-failure. There are two categories of failure. An inner-shell rupture would be a major incident, but the rupture of both primary shells would be a major incident of much higher consequence. References 1 and 2 make no claim for protection by the trip for low level of bulk sodium under these conditions.

c. Loss of secondary cooling capacity endangering the reactor or the primary tank.

The following discussion of these three possibilities indicates that the trips for bulk sodium are unnecessary.

7.6.3.1 IHX Rupture. The results of an analytical simulation of the primary and secondary systems after a complete tube-sheet rupture of the IHX are presented in Appendix B. These results indicate that 3500 gal (13.3 m^3) of secondary sodium would drain from the secondary system to the primary tank within 48 s. The bulk-sodium level would rise 11 in. (27.9 cm), with a brief maximum peak primary-cover-gas pressure of 6.0 psig (41.3 kPa gauge), during this interval.

An 11-in. (27.9-cm) increase in bulk-sodium level would not cause unacceptable damage to reactor components for the following reasons.

The primary sodium will not contact the primary-tank cover. A clearance of 4 in. (10 cm) would remain between the top cover and the primary-bulk-sodium surface, and the top cover and rotating plugs would not be subject to the thermal stresses that would result from contact of primary sodium with the top cover.

The primary tank will not receive stress damage as a result of the IHX rupturing. This statement is verified by applying the results of Appendix B to the formulas of Appendix D. The added weight of 3500 gal (13.3 m^3) of sodium in the primary tank would cause a difference of 0.34 psig (2.3 kPa gauge) in the tank head. This pressure, added to the transient peak pressure, indicates that the vessel bottom would be subject to a maximum transient pressure of 6.34 psig (43.7 kPa gauge). The value of stress obtained from the equations in Appendix D at this pressure is 16,365 psi (112.8 MPa). The allowable working stress is 14,700 psi (101 MPa). According to Section VIII of the ASME Code,[18] the yield strength is greater than 23,500 psi (162 MPa). Although the level determined by analysis indicates that the stress intensity may approach the ASME working limit for a short time, the calculated stress indicates that no deformation or damage would occur to the primary tank, and its service life would not be reduced.

An increase in bulk-sodium level to within 4 in. (~10 cm) of the top cover will not result in any damage due to overheating of the structural members. The top cover and associated structure is cooled by the shield-cooling system, which is composed of two loops, one to cool the top structure and the other to cool the biological shield. Also, flow between the loops can be adjusted to account for any imbalance in the heat loads. The total heat load to the system is about 506,000 Btu/h (146.7 kW), of which 130,000 Btu/h (37.7 kW) represents the heat loss from the top structure. During a hypothesized IHX rupture, which results in a reduction of the cover-gas thickness to 4 in. (~10 cm), the total heat load to the system will remain constant or more likely decrease (because of injection of cold secondary sodium). However, because of the decreased thickness of the cover-gas blanket, slightly more heat may be exhausted from the top cover. The structures above the tank and top cover are instrumented, with readouts in the control room. Should the top-cover sensors indicate rising temperatures, cooling can be adjusted to maintain temperatures below the design limit of 225°F (107°C). Adequate time for action is assured by the large heat capacity of the top structure.

Thus any damage to the top cover because of overheating in the event of a rise in sodium level during the hypothesized accident is highly unlikely.

With an IHX rupture, bulk-sodium temperatures might increase because of loss of secondary cooling. The most probable event would be that secondary and primary coolant would mix and thus mitigate the consequences. Neither high sodium level nor high bulk-coolant temperature influence the course of the other condition. The increased temperature would result only in a slightly increased pressure in the cover gas. This increase was calculated to be 6.2 psig (42.7 kPa). This results in a stress of 17,140 psi (118 MPa) which is still below the ASME yield strength of 23,500 psi (162 MPa) at 750°F (399°C). The temperature resulting from this hypothesized event (at peak pressure) would be less than 750°F (399°C).

The effect of the decreased thickness of cover gas on the consequences of the Design Basis Accident (DBA) as outlined in Ref. 1 has also been considered. A situation in which the decreased depth of cover gas would be a factor in accident mitigation would require an extremely unlikely event (IHX rupture) followed by the Design Basis Accident; this sequence would be in the ultra-hypothetical category. The effect of the reduced cover-gas thickness is considered here only for completeness.

According to the analysis in Ref. 1, the peak pressure in the tank after the DBA is a function of the interface velocity and the speed of sound in the argon cover gas. These in turn determine the Mach number. The pressure behind the reflected wave is then given as a function of the Mach number and the ratio of specific heats. The peak pressures reported in Ref. 1 were calculated by assuming that the shock wave contacts the top of the tank, and the calculation did not consider attenuation in the cover gas. Therefore, the depth of the cover gas was not a factor in this calculation. It did, however, enter in calculating duration of the pressure pulse.

To determine how depth of the cover gas being reduced from its normal 15 in. (38 cm) to 4 in. (10 cm) affects duration of the pressure pulse, the calculations of Ref. 1 were repeated for the decreased depth. It was found that the pressure-pulse duration would be increased by 6×10^{-5} s (0.00192 versus 0.00186). This difference is considered inconsequential and well within the uncertainty of the calculations.

In summary, a postulated IHX rupture could neither damage the primary vessel or reactor internals nor prevent the completion of a trip. Further, a trip would not mitigate the plant damage caused by an IHX rupture. Therefore, it is concluded that the anticipatory trip for high level of bulk sodium is unnecessary.

7.6.3.2 Rupture of Primary Tank. A primary-tank rupture is classified as an extremely unlikely event.[4] Because of the possible consequences of this rupture, the tank is redundantly constructed, with an inner and an outer shell. Rupture of the inner shell would be detected by the instrumentation monitoring for both the tank level and sodium leaks. (The contact sensors for sodium-leak monitoring are in the annulus between the primary-tank shells.) Both types of instruments have alarm annunciators in the control room. The incident would not endanger the reactor, because the primary-pump inlets would remain under sodium and primary flow would continue through the core. A trip could not mitigate the effects of an inner-shell rupture, and the incident could not inhibit control-rod trip action. It is therefore concluded that an anticipatory automatic trip on low level of primary bulk sodium is unnecessary.

The DBA is the only conceivable accident capable of causing the simultaneous failure of both primary-tank shells. In addition to causing failure of both tank shells, the DBA is also assumed to cause core disassembly. In Refs. 1 and 2, no credit is taken for the trip due to low level of bulk sodium during a DBA, and no evidence is available to support the possibility of a successful trip completion or to identify the need for a trip after a DBA. Thus the trip for low level of primary sodium is not considered necessary for plant protection.

7.6.3.3 Loss of Secondary Cooling. A maximum initial rate of heating for bulk sodium of 0.6°F/s (0.33°C/s), could occur during a loss of secondary cooling (see Appendix F). An analysis of this incident indicated that loss of secondary cooling without a trip for bulk sodium temperature high would have no adverse effects upon the reactor. With no operator intervention, the core would be at zero power and critical at about 777°F (414°C), and the fuel temperatures would not rise appreciably. The incident was also analyzed assuming inappropriate operator intervention; the operator was assumed to maintain reactor power. In this case, the reactor would be tripped in about 50 s at a bulk-sodium temperature of about 730°F (388°C) and a maximum fuel-surface temperature of about 1105°F (596°C). Neither of the loss-of-cooling incidents would cause excessive temperature of fuel or coolant.

A maximum temperature rise of bulk sodium of less than 100°F (38°C) is expected during loss of secondary cooling. The analysis of the additional thermal stresses generated during a rise from 700 to 800°F (371 to 427°C) in bulk-sodium temperature is given in Appendix E. The results indicate that the temperature rise would cause (1) a maximum combined stress of 15 690 psi (108 MPa) on the inner-vessel wall, (2) a maximum combined stress of 12 250 psi (84.4 MPa) on the outer-vessel wall, and (3) a maximum combined bending stress of 15 420 psi (106 MPa) in the radial beam of the primary tank.

The allowable primary-vessel working stress is 14 700 psi (101 MPa). This value was selected from Section VIII of the ASME Code.[17] At the date of design and construction, the ASME Section II Code for nuclear vessels did not exist. Appendix E refers to Section III of the ASME Code[18] to obtain a minimum yield stress of 43 200 psi (297.7 MPa). However, the use of Section VIII indicates a lower minimum value of 23 500 psi (162 MPa) for yield stress. The results of Appendix E indicate that, with the lowest value of minimum stress, the primary vessel will not be subject to structural damage during a loss of secondary cooling.

The information in Appendixes E and F indicate that a loss of secondary-cooling capacity would not damage the reactor or the primary vessel. The reactor is adequately protected against this incident without a trip for high bulk-sodium temperature.

The effects of transient high temperatures on the IHX were also considered in the analysis. The safety analysis concluded that the limiting overtemperature transient would occur during a loss of secondary cooling with inappropriate operator intervention. Temperature of the bulk sodium would increase 30°F (17°C) in 50 s.

The original design of the IHX provided for a normal steady-state temperature difference of 20.9°F (11.6°C) between the tube metal and inner shell for a transient thermal loading of 60°F (33.3°C) ΔT with a 200°F (111°C) temperature increase at the bottom tube sheet at the outer edge. Adding the 30°F (16.7°C) temperature increase to the normal 20.9°F (11.6°C) increase results in a 50.9°F (28.3°C) temperature difference between the tube and inner shell, which is below the design value of 60°F (33.3°C).

Additional analysis for the design of the replacement IHX met the requirements of Code Case 1331-5 for design and upset loading, including thermal transients. The limiting thermal transient used for the design subjected the IHX to a 140°F (77.8°C) temperature increase in 120 s--1.16°F (0.64°C)/s--which is more severe than the transient--0.6°F (0.33°C)/s--postulated for this event. The replacement IHX is specified to be identical to the original, except for design of several welds. Therefore, the conclusions above also apply to the present IHX.

7.7 Revisions to Practices and Documentation

7.7.1 Operations

Operating limits were changed to remove the requirements for a reactor trip due to out-of-limit level and temperature for bulk sodium. In addition, an upper limit to the subassembly outlet temperature, measured at full power, was specified because of the desire to minimize thermal effects to the plant; it is not required because of safety.

Operating instructions have been changed to reflect removal of the trips for bulk-sodium level and temperature.

Emergency procedures have been changed to require actions on receipt of alarm on high or low level or high temperature within the limiting requirements.

7.7.2 Administrative Controls

The administrative controls required for safe plant operation were not affected by this modification.

7.7.3 Hazard Summary Report

The following portions of Refs. 1 and 2 are superseded by this document

ANL-5719 (Ref. 1)	ANL-5719 Addendum[4]
Table VII	Table III Fig. 35

Table VII of Ref. 1 lists primary-tank bulk-sodium temperature too high or too low (primary-system item 13) and level too high or too low (primary-system item 14).

Table III of Ref. 2 lists bulk-sodium temperature high (item 39), bulk-sodium level high (item 40), and bulk-sodium level low (item 41) as abnormalities causing trips and alarm during reactor operation.

Figure 35 of Ref. 2 is a line diagram of the reactor trip circuit.

No claim is made in the Hazards Evaluation Section of Ref. 1 or 2 for protective action from the bulk-sodium trips.

7.7.4 Technical Specifications

The operational requirements for bulk-sodium level and temperature (see Table XII) have been included as limiting conditions for operation in the EBR-II technical specifications.

TABLE XII. Limiting Criteria for Emergency Procedures for Bulk-sodium Level and Temperature

Channel or Equipment	Plant Condition (for startup, power operation, or unrestricted fuel handling)	Limiting Action after Violation of Plant Condition	Minimum Configuration	Limiting Action after Loss of Minimum Configuration
Bulk-sodium level	The bulk-sodium level shall not be <293 in. (7.44 m) or >305 in. (7.74 m) above the bottom of the inner primary tank.	Anticipatory shutdown or termination of fuel handling	One channel operable (alarm)	Normal shutdown or termination of fuel handling
Bulk-sodium temperature (redefined as reactor inlet temperature)	The reactor inlet temperature shall be ≥580°F (304°C) and ≤740°F (388°C)	Reactor trip (manual) or termination of fuel handling	Two temperature channels operable	Normal shutdown or termination of fuel handling

7.8 Conclusion

The trips for temperature and level of primary bulk sodium are not required for adequate plant protection. The analysis first considered all plant incidents that could conceivably result in out-of-limit bulk-sodium conditions. From this compilation, those incidents of concern or importance were selected for comprehensive investigation. Finally, the effect of each incident upon the reactor was analyzed. On this basis, removal of the subject trips does not reduce plant safety.

Removal of these trips enhances plant availability by eliminating sources of spurious trips. Increased plant availability in turn increases plant efficiency, provides a more desirable irradiation facility, and decreases the possibility of subjecting the entire power plant to undesirable thermal transients.

This modification also removed variables of anticipatory trips in conformance with Sec. 3.5 of RDT Standard C16-1T.[4]

This modification did not adversely affect the safety of the EBR-II reactor plant during normal operation or under any accident conditions discussed in Ref. 1 or previous addenda.

8. EBR-II PLANT MODIFICATION WAF-5088: REMOVAL OF TRIP FOR REACTOR-BUILDING ISOLATION FROM REACTOR SHUTDOWN SYSTEM

J. I. Sackett, R. N. Curran, W. F. Booty, and N. L. Gale

8.1 Summary

Analysis has shown that the removal of the EBR-II reactor-building-isolation trip function from the shutdown system in no way compromises plant safety. This trip was found to be unnecessary for reactor protection and ineffective in mitigating the consequences of identified reactor faults.

This modification was completed in April 1976.

8.2 System before Modification

Figure 45 shows the affected section of the shutdown system before and after this modification.

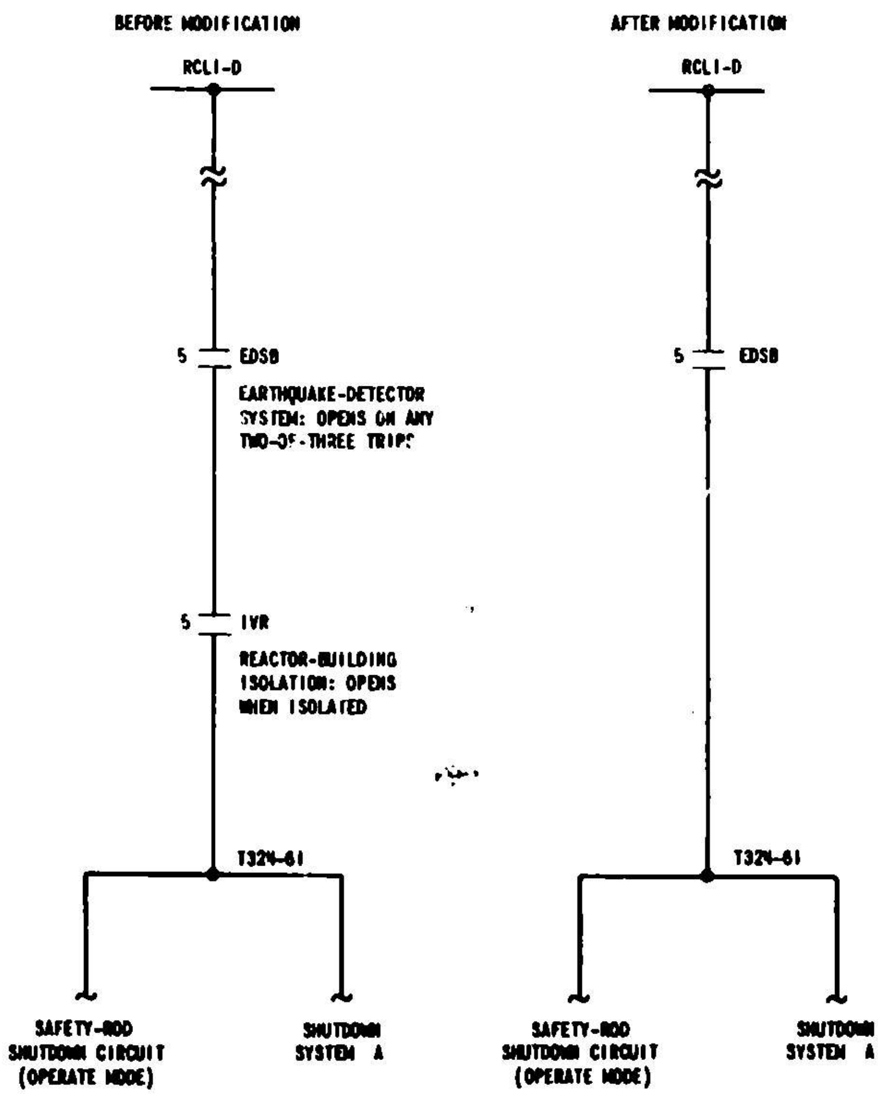


Fig. 45. Reactor-building-isolation Portion of Shutdown System before and after WAF-5088

The isolation system for the reactor building is tripped when any of the following conditions occur:

a. Excessive gamma radiation is detected at either of two locations on the reactor-building operating floor. The detector for RMS-20 (radiation-monitoring station) is on the top platform of the control-rod superstructure above the reactor. It is not fastened to anything on the platform, and the signal and power-supply leads are disconnected during fuel-handling operations to facilitate plug rotation. The electronics for this detector are at the bottom center of ICC-2.

The other detector is at RMS-21. It is adjacent to the inlet air blower on top of the personnel shield north of the personnel access door. The electronics are mounted on the shield below the detector. The normal setpoint for these instruments is 450 ± 50 mR/h ($1.2 \times 10^{-4} \pm 1.3 \times 10^{-5}$ C/kg·h).

b. Excessive pressure, with respect to ambient outside air pressure, is detected by a differential-pressure transducer on the outside wall of the reactor building, in the corridor near the personnel air lock. The normal setpoint for differential-pressure trip is 8 ± 2 in. H_2O (2 ± 0.5 kPa).

c. Excessive air temperature is detected in the reactor building by one of three temperature monitors, which are on the west wall of the reactor building, about 15 ft (4.6 m) above the operating floor. The normal setpoint for the three temperature monitors is 145 ± 5°F (62.8 ± 2.8°C).

d. Excessive subassembly outlet temperature is detected during reactor operation by two of four thermocouples. The four thermocouples are above subassemblies within the core (now in locations 2A1, 2C1, 3B1, and 3C1). The normal setpoint for the four trip channels is 110% of the temperature rise above the reactor inlet temperature. (These channels do not trip during fuel handling, but remain as functions for reactor trip and reactor-building isolation.)

8.3 System after Modification

The reactor trip on reactor-building isolation was removed for both the reactor-operate and fuel-handling modes by removing contact IVR-5 from the reactor shutdown string. One wire was installed, and unused wiring was removed. The annunciation of reactor-building isolation was removed from the trip section of the console annunciator. Annunciation of the reactor-building isolation is provided on the alarm section of the console annunciator. The parameters or trip logic of the reactor-building isolation system were not changed by removal of the IVR-5 contact from the reactor shutdown system.

8.3.1 Alarm Indications

The annunciator window on the trip section of the console annunciator that served the combined alarm function for trips for reactor-building isolation and earthquake detection was changed to indicate only earthquake-detection trips. An annunciator window for reactor-building isolation is on the alarm section of the console annunciator.

8.3.2 Calibration, Test, and Trip Setup

The test sources and calibration, test, and alarm-setup procedures for the reactor-building isolation system are unchanged by this modification.

8.4 Justification for Modification

The purpose of Plant Modification WAF-5088 is to upgrade the reactor shutdown system by:

a. Reducing the susceptibility of the plant to spurious trips by reducing the number of trip contacts without compromising plant safety.

b. Removing unnecessary anticipatory trips and thereby simplifying the reactor shutdown system, facilitating maintenance, reliability, and testing. (This is in compliance with the guidance of Sec. 3.5 of RDT C16-1T.[5])

The trip for isolating the reactor building is not needed. Trips for subassembly outlet temperature are included separately in the reactor shutdown system. The rest of the sensors in the building-isolation trip circuit do not respond to conditions within the primary system and are not related to its protection. The isolation trip is not needed to provide an anticipatory trip to ensure reactor shutdown before environmental conditions in the isolated reactor building reach a point at which reactor instrumentation would be damaged. There is adequate time for the operator to shut down the reactor after the receipt of alarms before the environmental conditions could so deteriorate that the reactor instrumentation would be adversely affected. Dependence on the operator to take appropriate action to prevent environmental conditions from exceeding operational conditions is allowed by Sec. 4.2.3 of RDT C16-1T.[4] Removal of this anticipatory trip function is therefore justified based on guidance in Sec. 3.5 of RDT C16-1T, which states that unnecessary anticipatory functions shall not be used.

Events originating within the primary tank, and leading to release of activity that should be minimized by reactor shutdown, are detected by instrumentation directly related to the primary system. For example, release of fission gas from single or multiple elements is directly monitored through measurement of cover-gas activity. Such activity would reach the containment building only after a long period as a result of slow leakage (about 1 L/min*) from the cover gas. Events that may lead to overheating of the reactor (and possible multiple-element failure) are protected against by reactor trips on

*This value is routinely measured by the responsible system engineer as part of a long-term effort to reduce cover-gas leakage.

over power, low flow, and high subassembly outlet temperature. A reactor trip on containment-building isolation would not reduce the activity released to the containment building under such circumstances.

Likewise, for the class of hypothetical events resulting in primary-sodium leakage from systems external to the primary tank, reactor trip would not significantly affect the amount of activity released to the containment building. Although shutting the reactor down under such circumstances (or terminating fuel handling) may be desirable, such action would not alter the course of the incident, and actions to be taken are covered in appropriate emergency procedures. (Instrumentation associated with the containment-building isolation would not detect such events. Such detection is more reliably made by smoke detectors, leak detectors, and local radiation monitors, for example.)

8.5 Applicable Standard

RDT C16-1T is the applicable standard--specifically, paragraphs 3.5, 4.3.4, and 4.3.6.

8.6 Safety Analysis

8.6.1 Potential Events and Related Safety Considerations

Safety considerations for the modification are:

a. The effectiveness of the reactor-building isolation trip in mitigating the consequences of reactor faults considered in Refs. 1 and 2.

b. Circumstances of building isolation that also require reactor shutdown.

EBR-II emergency procedures require anticipatory shutdown and termination of unrestricted fuel handling when the reactor-building temperature exceeds 100°F (38°C). Nuclear-thimble temperatures above 200°F (93°C) at chamber elevation also require reactor shutdown and termination of fuel handling, with shutdown to begin when temperatures reach 135°F (57°C).

None of the identified events has occurred. One spurious trip for reactor-building isolation occurred early in 1976.

8.6.2 Analytical Investigation

8.6.2.1 Detection of Abnormal Reactor Condition. A reactor trip from a building-isolation signal could be justified if one of the signals that causes isolation could detect an abnormal condition in the reactor and start protective action in time to prevent a reactor accident or mitigate the consequences of an accident. The only isolation signals that sense a condition inside the reactor are the thermocouples for subassembly outlet temperature, which are directly connected in the reactor shutdown system and are not removed by this modification.

Analyses in Refs. 1 and 2 show that any postulated accident (for either reactor operation or fuel handling) that could cause reactor damage progresses too rapidly for the signals for reactor-building pressure, temperature, or radiation to start a trip before the accident had run its course. Therefore, it is concluded that a trip from building isolation would not prevent or limit the consequences of the accidents considered in Refs. 1 and 2.

8.6.2.2 Effects on Building Temperatures of Reactor-building Isolation with Reactor Operating. Total isolation of the reactor building closes isolation valves in the Freon supply and return lines to the water-cooled condensers in the corridor adjacent to the personnel air lock. Thus, the refrigerant supply to the shield-cooling recirculation system is not cooled, with a resulting slow increase in reactor-building air temperature after total isolation. However, the increase is slow enough that it would take several hours before building air temperature increased to a level that could affect instrumentation of the PPS.

Experiment EBR-II EX-102 was conducted with the reactor shut down to determine the rate at which temperature increased at important locations in the reactor building with the service and ventilation system isolated for 2 h.

The heat loss from the primary system to the shield-cooling system is 430 000 Btu/h (124.7 kW)--300 000 Btu/h (87 kW) from the sides and bottom and 130 000 Btu/h (37.7 kW) from the top; only 15 000 Btu/h (4.35 kW) of the loss results from heating due to neutron and gamma attenuation. Thus, the reactor itself contributes a negligible additional heat load to the shield-cooling system. An additional 76 000 Btu/h (22 kW) is generated by the recirculating supply blower. The total of 506 000 Btu/h (146.7 kW) is equivalent to 20 000 cfm (9.44 m^3/s) of air at 12.2 psia (84.1 kPa) and 92°F (33.3°C) average with a temperature rise of 30°F (16.6°C). These temperature and flow rates are close to nominal operating conditions.

The only significant additional heat load inside the reactor building with the reactor at 62.5 MWt, instead of shut down, is from the secondary sodium system. This additional effect should not be particularly large. As noted above, the reactor top and structure contribute less than a third of the total heat load for shield cooling. Temperatures of the primary-tank top cover are about 360°F (182°C) with the reactor shut down and 400°F (204°C) at power. They are 50-100°F (28-56°C) hotter at power in the region around the IHX and secondary-sodium piping. The shield-cooling outlet temperature increases from about 104 to 109°F (40 to 43°C) at power. Data on thimble-cooling temperatures indicate essentially no effect due to reactor operation. Primary-tank sodium (and blower work) is the only significant heat source for the thimble-cooling system.

Thus, the test results of EX-102 should be reasonably applicable to isolation with the reactor at power. The following conclusions were drawn from the test data.

a. The building air temperature would not exceed acceptable instrument-operating temperatures of 120°F (49°C) during a 9-h period of isolation. This conclusion is based on an initial temperature of 75°F (24°C) and a rate of rise of 5°F (2.8°C)/h.

b. A limiting blast-shield temperature of 150°F (65.5°C) would not be reached during a 15-h isolation. This is based on an initial temperature of 95°F (35°C) and a rate of rise of 3.4°F (1.9°C)/h.

c. Temperature of instrument-thimble air would not reach the limit of 135°F (57.2°C)--the temperature at which a reactor shutdown is required by the technical-specification limits--until after the building had been isolated for 6 h. This is based on an initial temperature of 95°F (35°C) and a rate of rise of 6.3°F (3.5°C)/h.

Therefore, whether or not the reactor is at power, time is available for operators to manually shut the reactor down, if necessary, before reactor-building air temperatures increase significantly. This modification does not affect the existing alarm capability to warn the operators of temperatures higher than normal.

Also, after reactor-building isolation, the emergency argon supply is sufficient to maintain systems (such as the cover gas) at normal levels for a minimum of 15 days. [The emergency argon supply consists of eight argon cylinders containing a total of 45 000 L (STP). The total usage rate for all argon supply systems within the containment building is less than 3 000 L/day. Makeup to the cover-gas system is normally less than 700 L/day.]

8.6.2.3 Sodium Spillage with Concurrent Building Isolation. If sodium spillage were to occur, building isolation might result from an increase in building temperature. Removing the reactor trip function from the building-isolation system does not remove a potentially useful protective action for sodium spillage. Major sodium spillages are indicated more directly and reliably by smoke detectors, and automatic reactor trip is unnecessary. Emergency procedures outline the appropriate corrective action, and high building temperature is a marginal indicator for these events.

The temperature rise inside the isolated reactor building that could result from sodium burning after a large spill has also been analyzed. The worst-case spill requiring isolation is considered to be caused by a complete rupture of the primary-sodium purification loop, combined with a failure of the automatic vacuum-breaker system that is designed to stop purification-system flow when a leak is detected. The sodium pool would be limited to the surface area of 74 ft^2 (6.88 m^2) of the drip pan in the purification cell. Wi' data from the test reported in Ref. 19, the containment-building air tempe. ture (with the building isolated) was calculated to increase less than 0.4°F (0.22°C)/min from a pool burning area of 74 ft^2 (6.88 m^2). (See Appendix G.) Operators would be alerted by leak alarms within a few minutes and would have adequate time (more than one hour) to shut down the reactor, in accordance with the normal procedure for response to a sodium leak in the reactor building, before the air temperature could increase sufficiently to significantly affect reactor-shutdown instrumentation.

A leak from the secondary-sodium lines inside the reactor building could conceivably spill more sodium than the primary purification system. Such an extremely unlikely event would not require automatic containment-building isolation, however, because the low specific activity of the sodium results in negligible radiological consequences. [The entire secondary-sodium system contains only about 1 Ci (3.7×10^{10} Bq) of ^{24}Na. The worst-case activated sodium spillage that would require containment-building isolation is the hypothesized complete rupture of the purification cell, discussed previously.]

A brief discussion of a hypothesized spill of secondary sodium indicates how minor the problem is. The elevation of the secondary-sodium lines inside the reactor building is 5127.5 ft (895 m). A leak in a secondary-sodium line outside the primary tank at this elevation could eventually cause up to 1660 gal (6.3 m^3) of sodium to spill if no plant or operator action were taken. This volume consists of 560 gal (2.13 m^3) through the hot leg and 1100 gal (4.18 m^3) through the cold leg. However, we estimate that only a few hundred gallons could be spilled before many alarms would occur. For example, the alarm for low level in the secondary surge tank would occur after the surge tank had lost 160 gal (0.61 m^3).

Additionally, the secondary-sodium pump automatically shuts off if the surge tank loses 214 gal (0.81 m^3). These alarms provide a more positive indication of a leak in the system than any instrumentation associated with containment isolation. In addition to the alarms and protective action for secondary-system level, smoke detectors beneath the deck plates would provide early indication of a secondary-sodium leak. Upon verification of a valid fire alarm, the reactor plant will be manually tripped as required by emergency procedures, which could include, if appropriate, an emergency dump of the secondary-sodium system. This action could essentially be completed within 3 min after the shift supervisor performs the activation.

For the above reasons, the maximum possible spill of 1660 gal (6.31 m^3) of secondary sodium is extremely unlikely. However, in the most pessimistic situation, even if the maximum spill occurred, the resulting air temperature and pressure transient would not initiate isolation or challenge the containment-system capability; the ventilation system (if operating in the unisolated or partially isolated mode) would limit the increase in building-air temperature and pressure. The ventilation system in its normal operating mode removes about 500 000 Btu/h (145 kW). The heat load to the air from the hypothesized worst-case sodium burning in the primary purification system is 142 000 Btu/h (41.2 kW).

The worst-case secondary-sodium spill could result in a correspondingly higher heat load to the ventilation system; however, the system capacity is sufficient to keep air temperatures and resultant building pressure below values that would initiate automatic reactor-building isolation or, with the containment building partially isolated, significantly below design values of pressure and temperature. If building pressure should rise and radiological considerations do not require building isolation, pressure may be relieved manually as specified in Table XIII to allow personnel access or to prevent components within the building being damaged from high pressure.

The proper action to be taken upon sodium spillage depends on several factors, which include (1) limiting radiological release, (2) placing the reactor and associated system in a safe configuration, (3) facilitating efforts to limit or stop sodium burning, (4) maintaining cooling in the reactor building, and (5) controlling release of sodium oxide from the reactor building. Most of these functions are best served by a partial isolation of the containment building, an administrative action that allows building cooling to be maintained while release to the environment is limited. Alarms related to the need for reactor trip or shutdown are obtained from instrumentation in the vicinity of the affected area or from the leaking system itself. The proper action will be determined by the shift supervisor as guided by emergency procedures and his evaluation of the specific situation for the factors discussed above.

In summary, the existing system for containment-building isolation will not automatically actuate if sodium spillage occurs unless an associated activity release trips the radiation monitors. Without activation of the automatic isolation system for spillage of secondary sodium, the containment isolation system could not initiate automatic reactor trip. Therefore, removal of the reactor trip upon reactor-building isolation does not affect the safety of the plant for leaks of secondary-sodium piping within the containment structure. Protection of the reactor upon spillage of sodium, radioactive or nonradioactive, is provided by systems (smoke detectors, radiation monitors, or leak detectors) and procedures independent of the containment isolation system.

Reference 1 makes no mention of reactor trip upon building isolation. Reference 2 includes this trip in its description of the reactor shutdown system (Table III) and building-isolation system (Sec. A.7), but makes no claim in the hazards evaluation section (Sec. II, Appendixes F, G, and H) for protective action from this trip. Therefore, the safety of the plant is unaltered by this modification.

TABLE XIII. Limiting Criteria and Instrument Configuration for Parameters Related to Containment-building Environment

Plant Condition	Action after Loss of Condition	Minimum Configuration	Action after Loss of Minimum Configuration
Reactor-building temperature < 100°F (38°C); isolation trip at 140 ± 5°F (60 ± 2.8°C)	Anticipatory shutdown temperature ≥ 100°F (38°C)	One temperature channel operable. Alarm at 100°F (38°C); isolation trip at 140 ± 5°F (60 ± 2.8°C)	Normal shutdown; termination of fuel handling
Reactor-building pressure ≤ 8 in. H_2O (2.0 kPa); vacuum ≤ 8 in. H_2O (2.0 kPa); isolation trip at 8 ± 2 in. H_2O (2.0 ± 0.5 kPa)	Anticipatory shutdown and relief of pressure or vacuum subject to constraints for required containment	One pressure channel operable. Alarm and isolation trip at 8 ± 2 in. H_2O (2.0 ± 5 kPa)	Normal shutdown; termination of fuel handling
Radiation level satisfactory; isolation trip at ≤500 mR/h (130 μC/kg·h)	Restriction of personnel access, reactor-building evacuation, and/or reactor shutdown upon judgment of shift supervisor	Two radiation monitors operable (alarm and automatic isolation)[a]	Normal shutdown; all rods in least reactive position; termination of fuel handling
Reactor-building-exhaust gross release rate < 0.7 Ci/min (25.9 GBq/min) averaged over any 24-h period--may be waived, if approved by ERDA-ID and justified by one or more of the following:	Normal shutdown	One monitor available (alarm) in reactor-building stack exhaust	Normal shutdown; termination of fuel handling
		With cover-gas purge in operation, an operable monitor (alarm) on purge line or at main stack exhaust	Termination of purge
a. Identification of specific isotopes released			
b. Restriction of release to favorable atmospheric conditions			
c. Limiting duration and frequency of release			
Airborne particulate activity level within ERDAM 0524 limits [currently 3 fCi/mL (1.1 μBq/mL) with no alpha emitters present, 0.3 aCi/mL (110 mBq/mL) with unidentified alpha emitters present]	Restriction of personnel access, reactor-building evacuation, and/or reactor shutdown upon judgment of shift supervisor	One monitor operable (alarm)[a]	Restriction of personnel access; provision of portable monitoring
Gamma activity level within ERDAM 0524 (currently 5 Rem/yr or 3 Rem/calendar quarter in controlled areas)	Restriction of personnel access, reactor-building evacuation, and/or termination of fuel handling	As determined by the plant operations manager[b]	Restriction of personnel access; provision of portable monitoring

[a]One radiation monitor operable (RMS-21) during unrestricted fuel handling.
[b]These monitors are required any time personnel are in the reactor building.

8.7 Revisions to Practices and Documentation

8.7.1 Operations

Reactor trip due to containment-building isolation has been removed from EBR-II technical-specification limits.

Operating instructions have been changed to reflect removal of the building-isolation trip. New operating instructions have been prepared to cover operation of the reactor with the containment building completely isolated. Table XIII shows the limiting criteria for emergency procedures upon receipt of indication of abnormal conditions of building temperature and radiation level.

These criteria relate to off-normal conditions for which some operator action is required. The safety analysis presented in this report indicates that automatic isolation upon high building temperature or pressure is not required, but may be considered at best anticipatory. (This modification did not remove these isolation functions, however.) For example, as indicated earlier, it would be highly unlikely that local sodium spills would result in high temperature sensed in the containment building. Likewise, except for a severe explosion in the reactor building, the building cannot be significantly pressurized until after isolation. [Total exhaust for ventilation is ~6600 cfm (3.1 m^3/s).]

The only severe explosion hypothesized to result in concurrent radiation release is the hypothesized Design Basis Accident (DBA). The parameter by which isolation is accomplished earliest for that sequence of events is "high subassembly outlet temperature" within the reactor. In addition, the radiation-monitoring system alone should be sufficient to isolate the reactor building before significant activity is released. No fault has been identified for which automatic isolation on high building temperature or pressure is required; indeed, these isolation trips have not been shown to provide useful functions for identified faults.

The limiting criteria indicated in Table XIII for high containment-building temperature relate to limits for operation of electronic equipment. Isolation is not desirable at that temperature because it would negate the capability for cool reactor-building atmosphere. The limiting criteria indicated in Table XIII for manual relief of building pressure apply to a condition in which the building is already isolated. (Only under this circumstance would the containment building be pressurized because of an increase in temperature, either from the normal heat load or from sodium burning.) Relieving building pressure manually to facilitate personnel access when required and to limit the potential for damaging internal components is desirable when isolation is not required to limit radioactive release or personnel exposure.

Trips to isolate the containment building because of high temperature and pressure are not supported as required by current analysis, but have been included in Table XIII for completeness in representation of the system after this modification.

8.7.2 Administrative Controls

The administrative controls required for safe plant operation are not affected by this modification.

Existing procedures provide administrative controls adequate to protect workers during entry into the isolated building. The reactor may be operated with the reactor building partially isolated. This condition closes the ventilation valves, but does not trip the reactor. Removal of the total-isolation trip from the shutdown system would not permit reactor-building radiation levels to become any different from those possible under the partial-isolation mode of operation.

8.7.3 Hazard Summary Report

This document supersedes the following portions of Ref. 2:

Table III, items 65-68

Section IV.A.7 (last sentence)

Fig. 35 (Reactor Plant Isolation System)

8.8 Conclusion

This plant modification does not adversely affect the safety of the EBR-II reactor plant during normal operation or under any accident conditions discussed in Ref. 1 or previous addenda.

APPENDIX A

Dynamic Simulation and Analyses for Loss of Coolant Flow

The dynamic simulation and analyses described in this section are based on the EROS code;[20] only salient features are described here. The EROS dynamic simulation has been used in analysis of EBR-II rod-drop measurements[21] and, in comparison with actual occurrences of loss of coolant flow, has given good agreement with measured results. Detailed comparison of temperatures calculated by EROS and MELT-II for conditions of loss of flow agree well. The core model is based on configuration 56''' (essentially the core loading for run 56C), which was chosen as representing a well-characterized core and operation with the stainless steel reflector. Major items of importance to ensure applicability of such a core as a base case for kinetic simulation upon loss of flow (LOF) are that: the power and flow distributions be appropriate; nuclear parameters and reactivity-feedback characteristics be representative of current and future core loadings; and flow-coastdown characteristics be assumed conservative for subsequent operation. Limits have been developed for those parameters that affect behavior of the reactor and plant-protection system (PPS) upon loss of flow to ensure continued safe operation for varying core loadings. The parametric calculations on which those limits are based are presented in Sec. 5 of this appendix.

Nuclear data used for the calculations are shown in Table XIV.

TABLE XIV. Nuclear Data for EROS Simulation[a]

	Base Core	Reduced-feedback Model
Delayed-neutron fraction	0.0066	
Prompt-neutron lifetime	3.1×10^{-7}	
Maximum worth of one control rod, \$	1.3	
Maximum worth of two control rods, \$	2.6	
Prompt temperature coefficients, 10^{-3} \$/°F		
Core		
Uranium	-0.350	-0.142
Sodium	-0.680	-0.277
Axial reflector		
Sodium	-0.351	-0.143
Radial reflector		
Sodium	-0.122	-0.050
Total	-1.503	-0.612
Control-rod-bank expansion, \$/ft	2.926	2.926
Power-reactivity decrement calculated by EROS at 62.5 MWt		
Linear component, \$	-0.3022	-0.1435
Nonlinear component, \$	0.0362	0.0362
Total PRD, \$	-0.2660	-0.1073
Power coefficient, \$/MWt	-0.0043	-0.0017

[a]Conversion factors: \$/°C = 1.8 \$/°F; 1 ft = 0.3048 m.

1. Dynamic Simulation

In the dynamic studies described below, temperature computations were made with EROS, which continuously evaluates the closed-loop reactivity feedback as a function of detailed temperature changes with time, reflecting the effects of temperature-induced reactivity feedback back into the program as a modification to the system reactivity.

Calculating transient temperatures associated with LOF-related events requires the dynamic simulation of all characteristics of the reactor and PPS that directly or indirectly influence the time-temperature relation. Those parameters included in the EROS dynamic simulations that directly affect the first temperature peak are primary-coolant flow rate, reactivity feedback, and setpoint and response time of trip circuitry. Other parameters included, related primarily to the second temperature peak, are reactivity worth of trip rods, fission-product power, and auxiliary-pump flow.

2. Parameters Affecting First LOF Temperature Peak

2.1 Primary-coolant Flow Rate

The time-dependent data for coolant flow used in the EROS dynamic studies simulating loss of primary pumping power were data generated by analytical equations derived by fitting selected experimental data for flow coastdown to the appropriate exponential expressions. The original data, a combination of those taken during reactor trip on July 26, 1973, due to loss of flow and the flow-coastdown tests of May 1973, were corrected for dynamic measuring lag, flowmeter nonlinearity, and variation from design flow. These data represent the most rapid decrease in flow recorded to date and thus represent worst-case conditions.

Because EBR-II operates at 91% of its full design flow, both sets of data represent coastdowns from 91% flow. The FLOWDOWN code was used to obtain factors for extrapolation of the measured data to flow coastdown initiated from the true 100% flow level. The following equation, derived from the FLOWDOWN results, expresses the correction for the initial flow level:

$$F_{100} = 0.006 + 0.828F_{91} + 0.425F_{91}^2 - 0.547F_{91}^3 + 0.288F_{91}^4, \quad \text{for} \quad 0.05 \le F_{91} \le 1.0,$$

where

F_{100} = flow-coastdown data corrected for a trip initiated at the full flow level--9000 gpm (0.567 m^3/s)

and

F_{91} = flow-coastdown data for a trip initiated at 91% of full flow--8190 gpm (0.516 m^3/s).

The above equation is valid only for extrapolating flow from 91 to 100% for loss of primary pumping power and should not be applied to other cases.

Because no experimental data representative of pump-seizure events are available, the time-dependent flow data used in EROS simulations of pump seizure were abstracted from calculated FLOWDOWN data modeling coastdown behavior after seizure. The flow-coastdown data used in the EROS simulations are shown in Fig. 34; the normalized value of 1.0 corresponds to a flow of 9000 gpm (0.567 m^3/s).

Upon loss of flow due to any fault considered, fuel and coolant temperatures rise rapidly until the protective action (reactor trip due to low coolant flow or high subassembly outlet temperature) is initiated. The rate of temperature rise and the resultant temperature peak are thus directly related to the rate of coolant-flow decrease as it affects the power-to-flow (P/F) ratio.

2.2 Reactivity Feedback

During a flow coastdown, driver fuel, cladding, and coolant temperatures change throughout the reactor. As noted previously, temperatures increase promptly to an initial peak, decrease abruptly after a reactor trip, gradually increase again to a second peak, and finally decrease as power declines with constant flow. The effects of such changes are sensed directly as changes in the reactivity balance through various temperature coefficients of reactivity.

The EROS program includes a dynamic simulation of these effects. Basically, the program computes power as a function of reactivity change. Power changes affect temperatures, which in turn influence reactivity, which in turn influences power in closed-loop behavior. The program considers the continuously changing closed loop of power-temperature-reactivity-power through a model that divides the reactor into eleven generalized feedback regions for prompt temperature coefficients. The program computes node temperature for the fuel and sodium coolant as a function of time. The effects of temperature changes in these nodes are translated into reactivity through the appropriate temperature coefficients. The feedback network includes effects of control-rod-bank expansion and bowing. The resulting time-dependent system reactivity then modifies the reactor power.

Immediately after any LOF event considered, the degree of negative reactivity feedback (and corresponding loss of power) directly affects the rate of temperature rise. The time required to reach the setpoint for high coolant outlet temperature is also dependent on the magnitude of inherent reactivity feedback.

2.3 Setpoint and Response Time of Trip Circuitry

The PPS response times to all faults considered were simulated by the EROS[20] and CSMP[7] codes to establish a trip point consistent with the setpoint listed in Sec. 4.6.3. For the low-flow trips, flow-coastdown data were used as the input driving function to models of the flow monitors using the CSMP code. For temperature trips, transfer-function models of temperature instrumentation, based on the time constants given below, were programmed directly into EROS. The driving function for a temperature trip is the calculated coolant outlet temperature at the top of a driver-fuel subassembly in row 2.

Models used for calculation of instrument response are:

a. Monitors for High-pressure Plenum No. 2 and Total Flow

Time constants, s

MV/I input filter	1.15
MV/I	0.3
MV/I	0.3
Rochester trip (prefilter)	0.4

Delays, s

Control-rod clutch	0.020
Clark relays (2)	0.030
Rochester trip	0.200
	0.250

$$\frac{F_{out}}{F_{in}} = \frac{1}{(1.15S + 1)(0.3S + 1)^2(0.4S + 1)}.$$

b. Monitors for Low-pressure-plenum Flow

Time constants, s

MV/I input filter	1.15
MV/I	0.3

Delays, s

Control-rod clutch	0.020
Clark relays (2)	0.030
Rochester trip	0.200
	0.250

$$\frac{F_{out}}{F_{in}} = \frac{1}{(1.15S + 1)(0.3S + 1)}.$$

c. Monitors for Subassembly Outlet Temperature

Time constants, s

Thermocouple system	0.5
MV/I	0.178
MV/I	0.069
Rochester trip (prefilter)	0.5

Delays, s

Control-rod clutch	0.020
Clark relays (3)	0.045
Rochester trip	0.200
	0.265

$$\frac{T_{out}}{T_{in}} = \frac{1}{(0.5S + 1)(0.178S + 1)(0.069S + 1)(0.5S + 1)}.$$

Table XV summarizes the instrumentation response calculated by CSMP and EROS codes with models listed above.

TABLE XV. Instrumentation Response Calculated by CSMP and EROS Codes

LOF Protection Subsystem	Time When Control Rods Begin to Move Out of the Core	
Response of flow instrumentation to a step in flow from 100 to 0%	LPP flow monitor with trip levels at 85%: 0.67 s or 87%: 0.64 s	HPP 2 and total flow monitors with trip levels at 88%: 1.14 s or 94%: 0.94 s
Response of subassembly-outlet-temperature instrumentation to a step in temperature from 700°F (371°C) to 900°F (482°C)	Nominal startup setpoint 840°F (449°C): 1.24 s	Worst-case startup setpoint 847°F (453°C): 1.31 s
Response of subassembly-outlet-temperature instrumentation to a temperature ramp rates of:	Nominal at power setpoint; 110% of subassembly ΔT = 200°F (111°C)	Worst-case at power setpoint; 115% of subassembly ΔT = 200°F (111°C)
20°F (11°C)/s ramp[a,b]	2.04 s	2.56 s
110°F (61°C)/s ramp[a,c]	1.00 s	1.15 s

Summary of LOF Protection Subsystem of EBR-II PPS to identified LOF events	LPP Monitor Trip Level 85%	HPP #2 and Total Flow Trip Level 88%	Subassembly Outlet Temperature Trip Level 115% of Nominal Temperature Rise
One pump seizure and one pump coast-down (1.0 → 0.05)	1.24 s	1.74 s	1.59 s
One pump seizure and one pump operating (1.0 → 0.56)	1.24 s	1.74 s	1.59 s
Loss of primary pumping power at full flow (1.0 → 0.05)	3.19 s	3.52 s	3.26 s

[a]Input to MV/I; does not include thermocouple time constant.
[b]Ramp rate similar to a rate of temperature increase for an anticipated fault of loss of primary pumping power.
[c]Ramp rate similar to a rate of temperature increase for an unlikely fault of seizure of one pump and coastdown of the second.

3. Additional Parameters Used in EROS Calculations That Are Not Related to First Temperature Peak

3.1 Reactivity Worth of Trip Rods

For operation at full power, 4.08 $ of shutdown reactivity is normally vested in the control rods. This value was used to simulate faults with reactor trip.

After a reactor trip, the initial reduction in power (and corresponding delayed-neutron flux) is a function of the amount of negative reactivity insertion in a given time, i.e., trip-rod reactivity worth. Hence, after a reactor trip for any LOF fault analyzed, trip-rod worth will directly affect the P/F ratio, which, in turn, will affect the magnitude of the second temperature peak.

3.2 Fission-product Power

In the EROS program, under full-power steady-state conditions, 7% of total power is assumed to be generated by fission-product decay. During and after a reactor trip, when the ratio of fission power to fission-product-inventory power continuously decreases, EROS models the decay of fission products according to established analytical methods.[22]

After an LOF reactor trip, delayed-neutron effects will rapidly diminish; the main source of after-glow heat is then the residual fission-product inventory. Because by this time the flow is approaching a constant value, the fission-product power, determined by operating history and effective half-lives of the inventory, will influence the severity of the second temperature peak.

3.3 Heat-transfer Coefficients Used

The description of the EROS code in Ref. 20 includes a discussion of the methods used for development and use of heat-transfer coefficients. Those coefficients are generated at the steady-state full-power, full-flow conditions and held constant throughout the LOF calculation. The same film coefficients are used for full-flow and reduced-flow conditions for the following reasons:

a. The film coefficient changes only from a value of 7.2 (Btu/s)/ft^2·°F [(146.9 kW/m^2·K) at full flow to 5.8 (Btu/s)/ft^2·°F] (118.3 kW/m^2·K) at 5% flow for hottest driver-fuel pin used in the calculation.

b. The high thermal conductivity of sodium makes the heat transfer from the cladding to the sodium insensitive to the value of the film coefficient.

c. If the film coefficient used in EROS had been adjusted as a function of flow, the cladding temperatures calculated by EROS would not have been changed significantly. [The cladding temperature calculated at 5% flow would have been 0.7°F (0.39°C) higher if the lower value of film coefficient had been used.]

4. Analysis and Results

EROS calculations were made for faults selected from Table IX. Table XVI summarizes data from these calculations.

TABLE XVI. Summary of Results of EROS Calculations

Fault Class	Postulated Malfunction	Time to Flow Trip,[a] s	Time to Temp Trip,[a] s	Time to Driver-fuel Temp Limit of 1500°F,[b] s: w/o[c]	Time to Driver-fuel Temp Limit of 1500°F,[b] s: w[d]	Fuel-cladding Temp: First Temp Peak, °F: Flow Trip w/o[c]	Flow Trip w[d]	Temp Trip w/o[c]	Temp Trip w[d]	First Peak Protective Margin, °F: Flow Trip	Temp Trip
Anticipated	Loss of primary pumping power	3.52	-								
	Mark-IA			14.3	9.87	1200	1278	-	-	222	-
	Mark-II			12.8	8.63	1230	1313	-	-	187	-
Unlikely	One pump seizure	1.74	-								
	Mark-IA			∞	∞	1266	1355	-	-	145	-
	Mark-II			∞	∞	1297	1391	-	-	108	-
Unlikely	Loss of pumping power and failure of flow trips	-	3.26								
	Mark-IA			14.3	9.87	-	-	1192	1269	-	231
	Mark-II			12.8	8.63	-	-	1221	1303	-	197
Unlikely	One pump seizure with coastdown of second pump	1.74	-								
	Mark-IA			5.35	3.78	1290	1383	-	-	117	-
	Mark-II			4.80	3.21	1323	1421	-	-	79	-
Unlikely	One pump seizure and failure of flow trips	-	1.59								
	Mark-IA			∞	∞	-	-	1264	1353	-	147
	Mark-II			∞	∞	-	-	1296	1390	-	110
Extremely unlikely	One pump seizure, second pump coastdown, and failure of flow trips	-	1.59								
	Mark-IA			5.35	3.78	-	-	1283	1325	-	125
	Mark-II			4.80	3.21	-	-	1317	1414	-	86

[a]Time when control rod begins to move.
[b]Conversion factor: °C = (°F - 32)/1.8.
[c]Without uncertainties applied.
[d]With uncertainties applied.

5. Parametric Studies

Parametric analyses have been conducted to investigate effects of changes in important variables on performance of the PPS for loss of coolant flow. Those that apply to first peak temperatures and therefore to speed and accuracy requirements of the protective instrumentation are (a) changes in initial, absolute flow rate, (b) initial bulk-sodium temperature, (c) PPS trip time, (d) feedback reactivity, and (e) trip-rod worth.

5.1 Uncertainty in Initial Coastdown Rate

As the absolute flow level is increased, the initial rate at which the flow coasts down on loss of primary pumping power is increased slightly. (This has been predicted in analytical models and confirmed by actual tests of flow coastdown.) Thus, the most stringent requirement for response times of the instruments will occur with the highest initial, absolute, flow level. In recognition of this fact and with the possibility of increasing flow in the future [from 91 to 100% of 9000 gpm (0.567 m^3/s)], the curves used for flow coastdown have been based on those expected at the higher initial flow levels.

5.2 Bulk-sodium Temperature

Changes in initial bulk-sodium temperature will not affect flow-coastdown characteristics, and flowmeter response will be changed only slightly by flowmeter temperature coefficients. However, trips for subassembly outlet temperature will be affected. Table XVII summarizes system response. If bulk-sodium temperature drifts upward with no concurrent increase in reactor power by operator actions, reactor power will be reduced by feedback effects, with a resulting decrease in subassembly outlet temperature. The reactor would become subcritical when the inlet temperature reached 800°F (427°C), but this condition could not occur if the heat-rejection system continued to operate normally. A decrease in bulk-sodium temperature will result in a trip on high reactor power when the sodium temperature has decreased by about 12°F (6.7°C).

TABLE XVII. Effect of Changes in Bulk-sodium Temperature on Response of PPS

Change in Bulk-sodium Temperature	Change in Power Level	Limit of Change	Change in Subassembly Outlet Temperature	Limit of Change	Change in Time to Trip on Loss of Flow: Flowmeters	Change in Time to Trip on Loss of Flow: S/A Outlet Temp
			No Operation Action			
A. Decrease	Increase	Trip at ≤15% increase in power level	Increase[a]	Limited by power-level trip to 30°F (17°C) increase	No change (protective margin decreased)	Reduced (protective margin essentially unchanged)
B. Increase	Decrease	Reactor would be subcritical at bulk-sodium temperature of 800°F (427°C)	Decrease[a]	Bulk-sodium temperature limited to ≤100°F (56°C) decrease by reactivity decrease	No change (protective margin increased)	Increased (protective margin essentially unchanged)
			Operator Action Maintaining Reactor ΔT at 183°F (102°C)			
C. Decrease	No change		Decreased by amount equal to change in bulk-sodium temperature	Limited to ≤120°F (67°C) decrease by heat-dump system	No change (protective margin increased)	Increased (protective margin slightly reduced)
D. Increase	No change		Increased by amount equal to change in bulk-sodium temperature	Limited to ≤30°F (17°C) increase in outlet temperature by trip for S/A outlet temperature	No change (protective margin decreased)	Decreased (protective margin slightly increased)

[a]Magnitude of change depends upon magnitude of isothermal temperature coefficient relative to power coefficient of reactivity. With current values, a 1°F (0.56°C) decrease in bulk-sodium temperature would result in a typical increase of 1.5°F (0.8°C) in subassembly outlet temperature.

In the event of a loss of flow concurrent with either of these conditions, protection is provided by flowmeters and sensors for subassembly outlet temperature. Temperature at the first peak is a monotonically increasing function of reactor power as well as bulk-sodium temperature. Thus, although an increase in power raises first-peak temperature, a decrease in bulk-sodium temperature would lower it. The net effect, with no change in time to trip assumed, is to limit to only slight changes the first peak temperature reached. Under no conditions could it increase by more than 30°F (17°C) for case A. For case B, the reduction in power-to-flow ratio would more than compensate for the increase in inlet temperature and would increase the protective margin for the flow trip. Because trips for subassembly outlet temperature are measuring outlet temperature, trips from those sensors occur earlier in the transient when the net effect of an increase in power and bulk-sodium temperature is to increase temperature.

The reactor operator controls change in reactor temperature (reactor ΔT) and flux level as primary variables. With the assumption that he maintained reactor ΔT at 183°F (102°C) (and reactor power at 100%) during changes in bulk-sodium temperature, the first change in peak temperature on loss of flow would correspond to the change in bulk-sodium temperature (assuming no change in time to trip). The increase in first peak temperature is limited to 30°F (17°C) by a trip on high subassembly outlet temperature. Because this trip would occur earlier in the loss-of-flow transient, resulting first-peak temperatures would be reduced somewhat from the corresponding increase in bulk-sodium temperature. Thus, only minor changes in first-peak temperatures for loss of flow result from changes in bulk-sodium temperature.

5.3 Sensitivity of Results to PPS Trip Time

Magnitude of the first temperature peak is determined by PPS setpoint and response, system feedback reactivity, and the change in rate of flow. Limits on driver-fuel temperature determine the time at which the reactor must be tripped, and the PPS-trip setpoint and response time must be capable of providing protection for all LOF transients. Shown in Fig. 46 is the highest cladding temperature for Mark-II driver fuel at the first and second temperature peaks for base-case nominal feedback reactivity and the reduced-feedback model (Table XIV) as a function of time at which the reactor is tripped. For feedback reactivity, the first peak cladding temperature increases by about 130°F (72°C) for a delay in time to trip of 3.75 s upon LPPP. By contrast, for the same case, the second temperature peak would increase by only 12°F (7°C), for a trip time of 3.75 s compared with a trip with no delay upon LPPP.

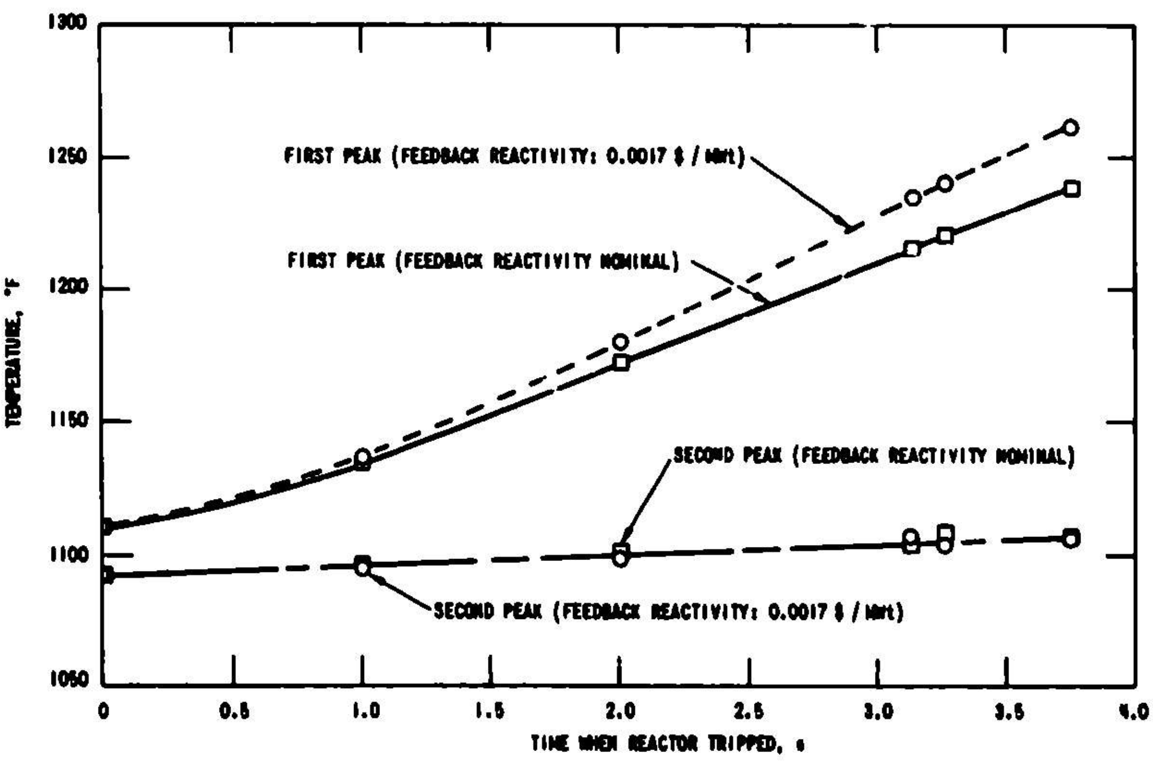


Fig. 46. Loss of Primary Pumping Power: Hottest Cladding Temperature for Mark-II Fuel for First and Second Temperature Peak as a Function of Time When Reactor Tripped: Shutdown Reactivity, 4.08%; Auxiliary Flow, 5.0%. Conversion factor: °C = (°F - 32)/1.8.

Calculations were also performed for an assumed decrease in flow to 3.3% (Fig. 47). Again, magnitude of the first temperature peak is a function of reactor trip time, whereas the second temperature peak is only slightly affected by trip time.

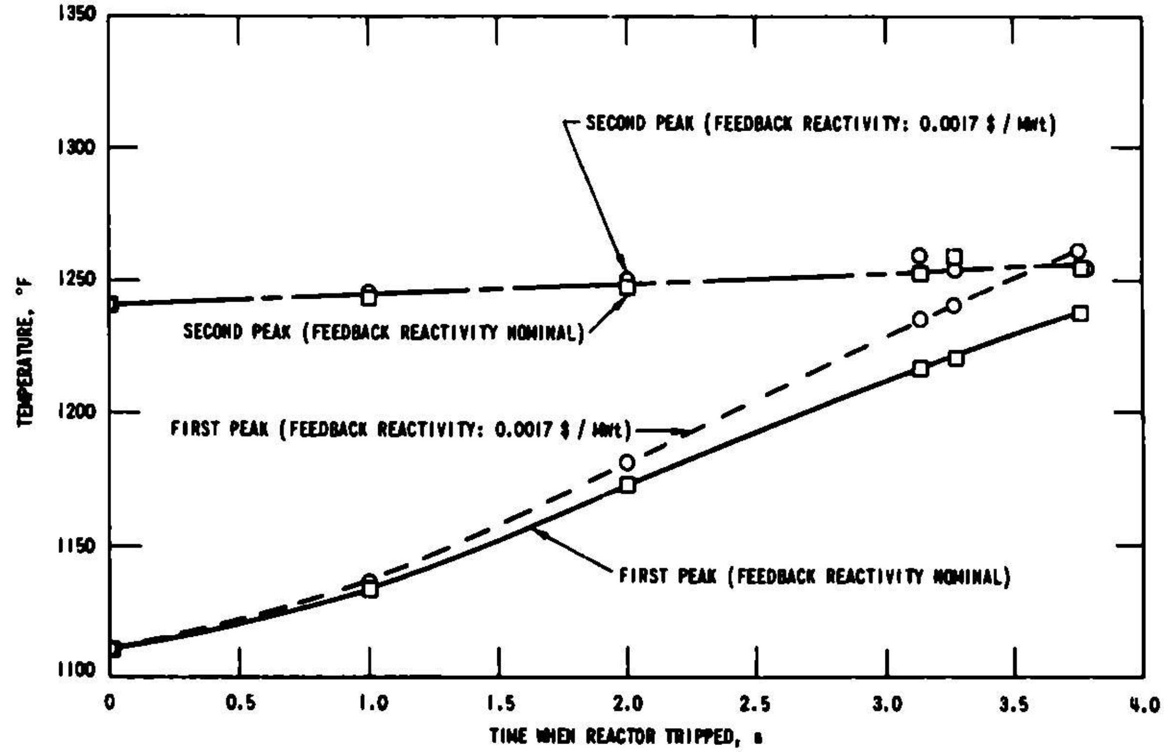


Fig. 47. Loss of Primary and Auxiliary Pumping Power: Hottest Cladding Temperature for Mark-II Fuel as a Function of Time When Reactor Tripped: Shutdown Reactivity, 4.08%; Residual Flow, 3.3%. Conversion factor: °C = (°F - 32)/1.8.

5.4 Sensitivity to Feedback Reactivity

Because feedback reactivity is important in determining temperatures upon LPPP, calculations were also performed in which feedback reactivity was assumed to be reduced 60% from nominal, the limit contained in EBR-II technical specifications. Figure 47 shows that at the reduced feedback, first peak temperatures upon LPPP are only 27°F (15°C) higher than for nominal feedback reactivity for a reactor trip time of 3.75 s. Second peak temperatures show no change.

For an LPPP with nominal feedback reactivity, a temperature trip will occur at 3.26 s; the resulting first peak temperature is 1221°F (660°C) without uncertainties. For an LPPP with the reduced-feedback-reactivity model, a temperature trip will occur at 3.13 s, with a resulting first peak temperature of 1235°F (668°C) without uncertainties. Thus, temperature trips would occur earlier in the transient as feedback reactivity is reduced to the limit and would limit the rise in first peak temperature to 14°F (8°C).

Calculations were also performed for an assumed reduction in flow to 3.3% with the reduced-feedback model. The results are shown in Fig. 47. The same conclusions apply: Magnitude of the first temperature peak is a function of feedback reactivity, but that of the second temperature peak is not.

5.5 Effect of Delay in Reactor Shutdown on Second Temperature Peak

Calculations have been performed to determine the sensitivity of the second temperature peak to delays in reactor shutdown after loss of pumping power. These results are also shown in Figs. 46 and 47. For time delays between 0 and 4.0 s, the magnitude of the second peak increases about 3.7°F (2°C). Therefore, the uncertainty in trip time, between 0 and 4 s, does not significantly affect the magnitude of the second temperature peak.

5.6 Effect of Trip-rod Worth on First-peak Temperature

Table XVIII summarizes the sensitivity of trip-rod worth on the first temperature peak for loss of pumping power for trips at 3.75 s with nominal feedback reactivity, and at 3.13 s with reduced feedback reactivity. Results show no change in first-peak temperatures as shutdown reactivity is changed.

TABLE XVIII. Effects of Shutdown Reactivity on First Temperature Peak during Transient due to Loss of Pumping Power

	Highest Mark-II Driver-fuel Cladding Temperature at First Temperature Peak					
	At Nominal Feedback Reactivity			At Technical Specification Limit for Feedback Reactivity		
		Temperature			Temperature	
Shutdown Reactivity, $	Time Tripped, s	°F	(°C)	Time Tripped, s	°F	(°C)
2.0	3.75	1238	(670)	3.13	1235	(668)
2.78	3.75	1238	(670)	3.13	1235	(668)
3.13	3.75	1238	(670)	-	-	-
4.08	3.75	1238	(670)	3.13	1235	(668)
5.0	3.75	1238	(670)	3.13	1235	(668)
		Normal feedback reactivity			0.0042 $/MWt	
		Proposed limit on feedback reactivity			0.0017 $/MWt	

5.7 Uncertainty in Calibration of Reactor Thermal Power

Uncertainty in trip settings of channels for subassembly outlet temperature and in establishing a power-to-flow ratio of 1.0 that corresponds to design calculations may be related directly to the uncertainty in accurately measuring reactor thermal power at 30 MWt and again at 62.5 MWt. A basic element in determining reactor thermal power is the reactor mixed mean outlet temperature as measured at the outlet plenum. An error in that reading would be reflected proportionally in the thermocouple-channel trip settings and the proper power-to-flow ratio. For protective action, an "unsafe" error would be one in which the reactor outlet temperature as indicated was lower than the actual temperature. As a check on this circumstance, temperatures in the secondary sodium system are compared with primary-system temperatures on startup and at power.

The most positive check on reactor outlet temperature is the outlet temperature of secondary sodium from the IHX. Obviously, this temperature cannot be higher than the reactor outlet temperature. At full power, secondary-sodium outlet temperature is 13°F (7°C) lower than reactor outlet temperature. Comparisons between outlet temperatures of primary and secondary sodium provide assurance that temperature readings are consistent and that absolute errors in calibration are well within 5% of total temperature rise.

The absolute accuracy of the flowmeter used to calibrate thermal power is of little importance, because a flow lower than that indicated would be compensated for by a corresponding reduction in the power level, and an operating point at a power-to-flow ratio of 1.0 would be correctly established to obtain the design values of fuel, cladding, and coolant temperatures in the hottest subassembly. Flow rates to be used during a reactor run are established before reactor operation. The power-range nuclear instruments are calibrated during reactor startup at about 30 MWt and again at full power. Thus, the error in power-to-flow ratio for any reactor run is sensitive only to the absolute accuracy of the temperature measurements made during the calibrations at about 30 MWt and again at full power.

This calibration uncertainty affects initial conditions assumed for reactor transients and is within the power-level factor (actually a power-to-flow factor) included within the uncertainty factor of 1.157 applied to calculation of fuel and cladding temperature. The uncertainty in calibration would not affect setpoints of protective instrumentation relative to initial conditions.

5.8 Uncertainty in Calibration of Subassembly Outlet Temperature

An important element of protective capability of subassembly outlet thermocouples is that they at all times reliably indicate temperature relative to mean reactor temperature rise. After a loading change, the subassembly outlet temperatures are uncalibrated until reactor power is about 30 MWt, at which time a calibration is made. The result is that the uncertainty associated with trip settings for subassembly outlet temperature is greater during startup than during full-power operation.

To ensure that during startup after a loading change the trip setpoints of these thermocouples are not greater than 115% of temperature rise at full power, initial setpoints have been established that are well below that level. To accomplish this, subassembly outlet temperatures were analyzed to obtain maximum and minimum temperatures for a variety of core loadings. Trip points were then established that are above maximum values obtained at 30 MWt, but below minimum values at 62.5 MWt.

After calibration at 30 MWt, trip settings for subassembly outlet thermocouples are extrapolated (linearly) to values corresponding to 110% of full power. At full power, the settings are checked to ensure that trip values are 110% of temperature rise corresponding to full power.

5.9 Uncertainty in Flowmeter Calibration

Electromagnetic flowmeters in the plant-protection circuit are calibrated by using a Foster flow tube (a ΔP flowmeter) measuring total reactor flow. Errors in absolute flow rate as measured by this flowmeter are therefore translated directly to flowmeters in the PPS. Absolute flow rate has been determined to be about 9% below the levels indicated by the Foster flow tube.

For action by the PPS, however, absolute flow accuracy is of minor importance. What is important is that trip settings relative to indicated full flow be accurately established. The parameter of significance to core temperature, and therefore severity levels, is power-to-flow ratio. With the mode of operation at EBR-II, this ratio is the controlled variable. [Reactor ΔT is limited to 183°F (102°C).] The nominal trip setting of the flowmeters for total flow and the high-pressure plenum is for a 6% reduction in flow (corresponding to a 6.4% increase in power-to-flow ratio and therefore ΔT). The worst-case trip point of 12% reduction in flow is at a 13.6% increase in power-to-flow ratio and allows for a 6% uncertainty in the trip point of these flowmeters--well within the accuracy capabilities of the instruments (see Table XVII).

Flowmeters for the low-pressure plenum have a worst-case trip point established at a 15% reduction in flow (corresponding to a 17.6% increase in power-to-flow ratio). Although this flow is lower than that for the flowmeters for total flow and the high-pressure plenum, their faster response results in trip times that compare favorably with those flowmeters for identified flow-coastdown incidents. For a normal trip-point setting of 88% of normal flow, the instrument accuracy is adequate to ensure a worst-case trip point of 85% of normal flow (see Table XVII).

An increase in bulk-sodium temperature to 715°F (379°C) during operation at full power is considered an upper limit to be expected during the plant lifetime. The trips for subassembly outlet temperature are normally set in a range where a 15-30°F (8-17°C) increase in bulk-sodium temperature would cause a reactor trip.

5.10 Definition of Uncertainty Factor

The uncertainty factors applied to calculated temperatures are essentially those used in Ref. 2, with one change. The factor for a 5% transient overload was applied as a multiplier rather than in a square-root-of-the-sum-of-the-squares technique. The basis of a statistical approach to uncertainty factor, rather than a strictly multiplicative one, is the assumption that no point will be subject to all the uncertainties at the same time. Therefore, applying the power-level uncertainty factor statistically would be inappropriate for events in which the entire core would participate.

The effect attributable to the overpower transient was removed from the calculation in Table VIII of Ref. 2, and an uncertainty factor of 1.102 was obtained. (Uncertainty factor is the ratio of temperature rise from inlet coolant to point of interest with and without uncertainty.) Application of the 5% power uncertainty as a multiplier to the uncertainty factor of 1.102 results in an uncertainty factor of 1.157. This uncertainty in power is considered to apply to the uncertainty in absolute power-to-flow ratio used in the initial conditions applied to transient calculations.

5.11 Whole-core Response to Unprotected LPPP Event

Because of the distribution of power and coolant flow across the core, not all subassemblies reach the temperature for formation of a fuel-cladding eutectic at the same time in an LOF transient. All calculations establishing safety-related limits are based upon the hottest subassembly in the core having an element with a coolant temperature at its top of 1040°F (560°C) for Mark-IA fuel or 1050°F (566°C) for Mark-II fuel.

To provide a measure of the degree of conservatism in that approach, calculations were performed to determine, element by element, the time at which a given percentage of core was approaching the temperature for eutectic formation. Shown in Fig. 48 is the time at which a given percentage of the core reaches eutectic temperature upon an LPPP transient, with no trip assumed. For the figure, it is assumed that 59 subassemblies of Mark-II fuel are distributed throughout the core and all elements are at 2.9 at. % burnup. Applying the uncertainty factor shows that the first fuel element, out of 5369, reaches the temperature for eutectic formation at 3.70 s after the LPPP (6.2 s without application of the uncertainty factor). Ten percent of the core is involved at 5.6 s and 50% of the core at 8 s.

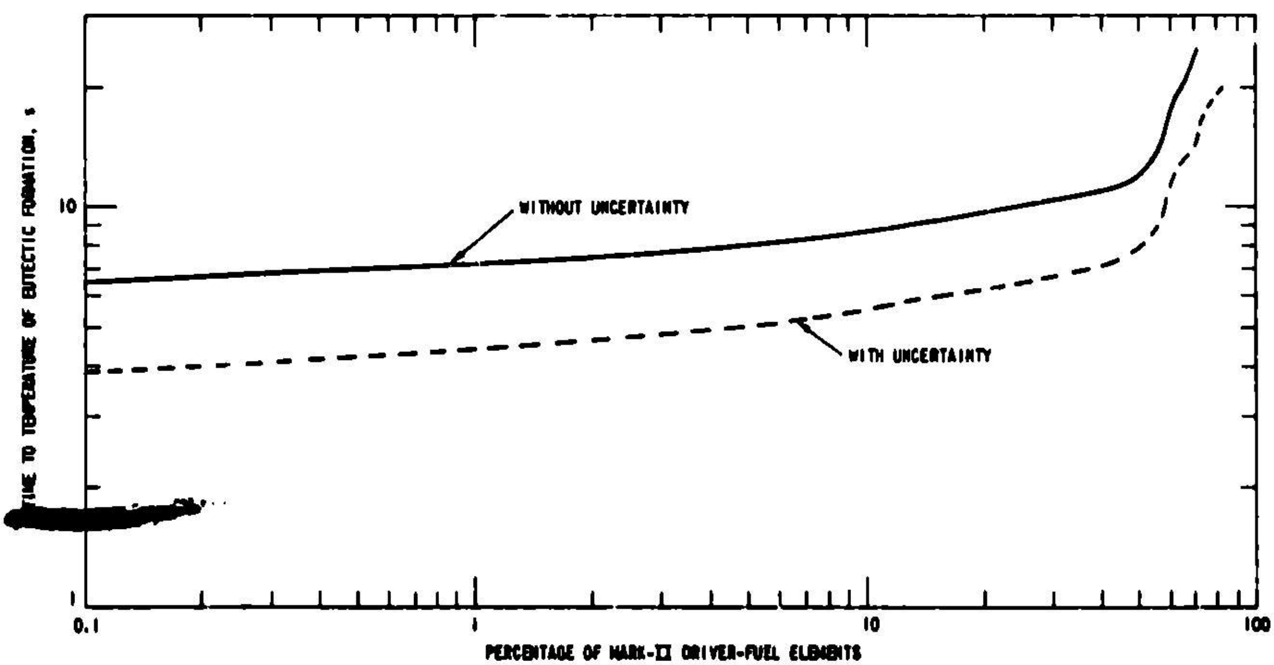


Fig. 48. Time for Mark-II Driver-fuel Elements to Reach Temperature of Eutectic Formation in an Unprotected Transient Caused by Loss of Primary Pumping Power

APPENDIX B

Effect of Secondary Sodium System on Primary Tank after IHX Rupture

1. Summary

Calculations indicate that, after a massive rupture of the IHX secondary piping, (a) adequate cover-gas space remains in the primary tank to contain the volume of secondary sodium expelled, and (b) peak pressures would not overstress the inner-vessel bottom. With the pressure-relief system operable, the bulk-sodium level in the primary tank would rise 11 in. (28 cm) in 42 s [15 in. (38 cm) is available] and the cover-gas pressure would peak at ~6 psig (41.4 kPa gauge) after 33 s.

2. Discussion

The function of the intermediate heat exchanger (IHX) is to provide heat transfer between the primary and secondary systems of EBR-II. The IHX is the only point of communication between the primary and secondary systems. The secondary system is maintained at a greater pressure [7 psig (48 kPa gauge) + elevation differences] than the primary system [~1 in. H_2O (249 Pa)] so that any possible leakage in the IHX would result in flow of secondary sodium into the primary tank rather than flow of primary sodium from the reactor into the secondary system and out of the containment building.

The incident leakage of secondary sodium into the reactor primary tank is of concern from the standpoint of (1) causing the level of primary-tank bulk sodium to rise and contact the top cover of the primary vessel and (2) causing excessive pressures in the primary tank. If the bulk sodium were to contact the primary-vessel top cover, deformation of the top cover due to excessive ΔT across it may be possible. Deformation of the top cover could cause control-rod binding and prevent reactor trip. Excessive pressures in the primary tank could cause or contribute to primary-tank failure.

Figure 49 is a flow diagram of the secondary system with respect to the IHX. This figure presents a functional look at the components of the secondary system that are included in this analysis. Secondary sodium is pumped from the surge tank to the IHX by the main electromagnetic pump. The heated secondary sodium then flows from the IHX through the superheaters and evaporators and back to the surge tank.

The initial conditions assumed for the accident study are as follows:

a. Before the accident, the reactor is at full power in normal operation. Significant parameters are:

Secondary flow rate	5794 gpm (0.37 m^3/s)
Storage and surge-tank pressure	7 psig (48 kPa gauge)

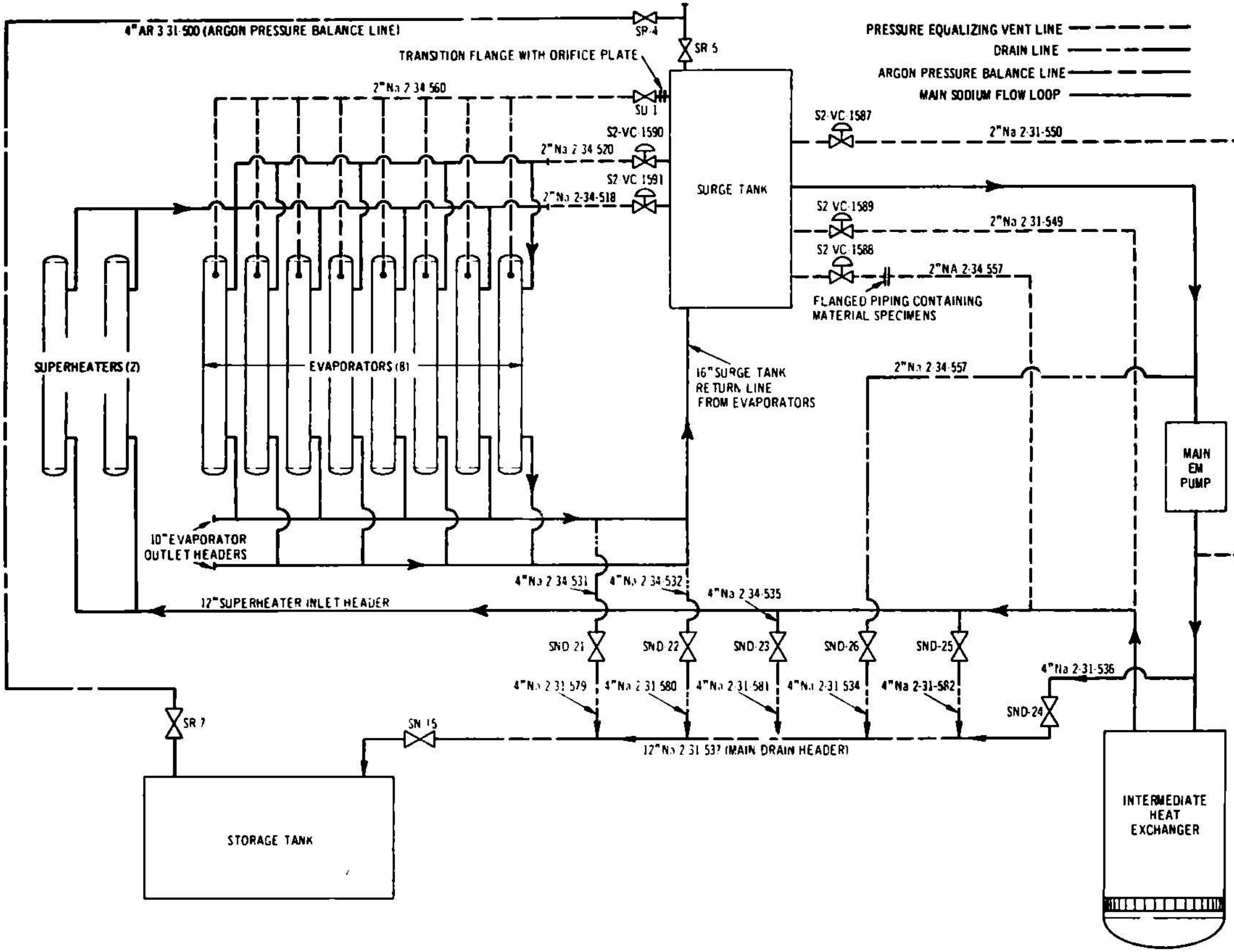


Fig. 49. Secondary Sodium System of EBR-II. Conversion factor: 1 in. = 25.4 mm.

Storage and surge-tank temperature	585°F (307°C)
Average hot-line temperature	800°F (427°C)
Cover-gas pressure in primary tank	1 in. H_2O (249 Pa)
Primary-tank altitude	5117.5 ft (1559.8 m)
Pressure-relief system of primary tank	Operable

b. The IHX undergoes a guillotine failure at the top of the cover such that:

(1) There is no resistance to sodium flow from the cold leg into the primary tank or to cover-gas flow into the hot return leg.

(2) Communication between the cold and hot legs is completely severed.

The discharge flow rate of the secondary pump remains at the initial value for a short time, even though flow is diverted and pump discharge impedance is reduced. This is because the pump is in flow-rate control. There would, of course, be oscillations due to the upset. This flow rate will be maintained until the surge tank has drained 30 in. (76 cm), from the normal 132.5-in. (337-m) level down to the 102.5-in. (260-cm) pump-trip level.

After the pump shuts off, the driving forces in the cold leg will be sodium inertia, the pressure differential between the primary and surge-tank cover gases, and the gravity head. The driving forces in the hot leg are similar.

Neither manual dumping of the secondary sodium nor manual relief of pressure of secondary-sodium cover gas is considered because of the very short time of the incident when compared to possible operator reaction time.

3. Analytical Method

The general approach taken was to repeatedly calculate parameters of the secondary sodium system during the interval from initiation of the above-defined accident until the system attained approximate equilibrium. The interval between calculations was made small enough to produce insignificant step changes in the variables. Briefly, each calculation consisted of:

a. Calculating the liquid-sodium flow rates in the pump (cold) and heat-exchanger (hot) lines.

b. Adjusting the various liquid and gas volumes, sodium heights and lengths remaining in lines, etc., for the time interval. The resulting new ullages (cover-gas volumes) in the storage, surge, and primary tanks produced new pressures.

c. Calculating the argon flow rates between the various ullages on the basis of the new differential pressures.

4. Analytical Results

Several off-normal cases were studied to determine the effects of various parameters on system performance: initial secondary flow rates of 10 and 1% of normal; surge-tank pressure of 6 psig (41.4 kPa gauge); resistance of the primary-tank exhaust line reduced by a factor of 10; and failure of the pressure-relief system to operate during the incident.

The only significant effect in all but the last case is a reduced peak primary-tank pressure. Of couse the shapes of the flow and velocity curves differ, especially at first, but the head term soon dominates and causes the later portions to behave similarly. The ultimate bulk-sodium level is the same in all cases.

In the last case, which assumes no pressure relief, gas pressure peaks at a higher value, 9.6 psig (66.2 kPa gauge), still well within the stress limits of the primary tank (see Appendix F). Bulk-sodium volume and level are correspondingly lower [320-ft^3 (9.1-m^3) increase in volume and 8-in. (20-cm) increase in level]. Again, the transient requires 41 s.

Figures 50-57 show the important variables calculated for 50 s of the full-power case. The pump line is the cold line from the side of the surge tank to the IHX, and the evaporator line is the hot line from the bottom of the surge tank through the superheaters to the IHX. Time is from initiation of failure.

Sodium in the surge tank drops to the trip level [30 in. (76 cm) below the normal operating level] after 5 s (Fig. 50). This drop trips the secondary pump, and the cold-leg flow rate (Fig. 51) decreases to 2800 gpm (0.18 m^3/s)

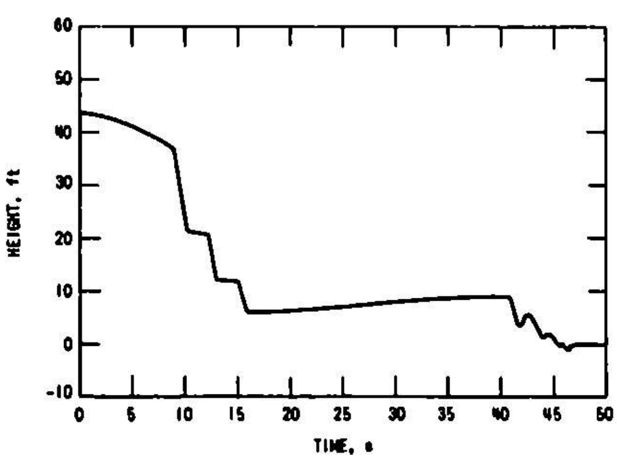


Fig. 50. Sodium Height in Pump Line above Primary Sodium. Conversion factor: 1 ft = 0.3048 m.

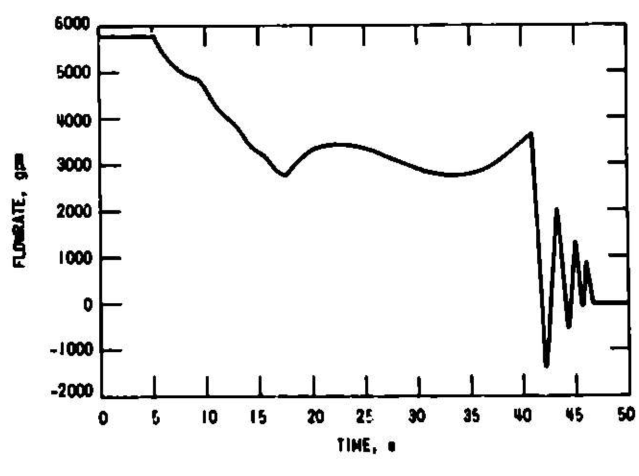


Fig. 51. Sodium Flow Rate in Pump Line. Conversion factor: 1 gpm = 6.309 x 10^{-5} m^3/s.

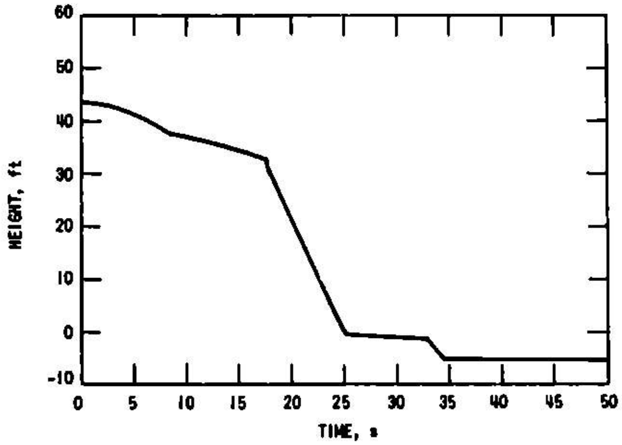


Fig. 52. Sodium Height in Evaporator Line above Primary Sodium. Conversion factor: 1 ft = 0.3048 m.

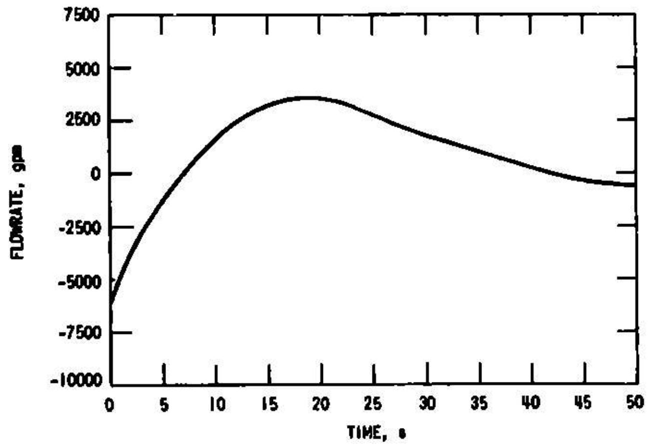


Fig. 53. Sodium Flow Rate in Evaporator Line. Conversion factor: 1 gpm = 6.309×10^{-5} m^3/s.

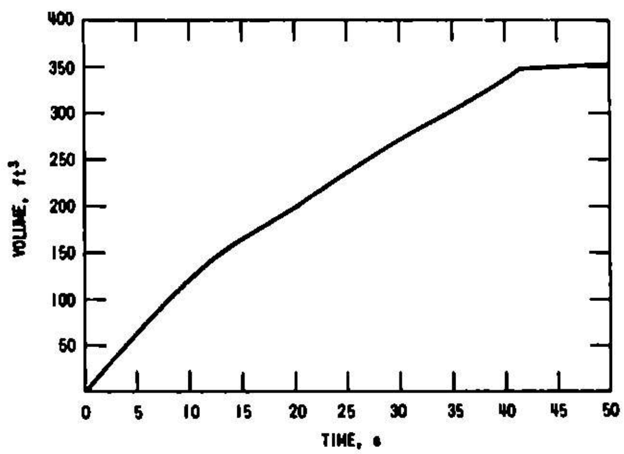


Fig. 54. Sodium Volume, Pump Line into Primary Sodium. Conversion factor: 1 ft^3 = 0.028 m^3.

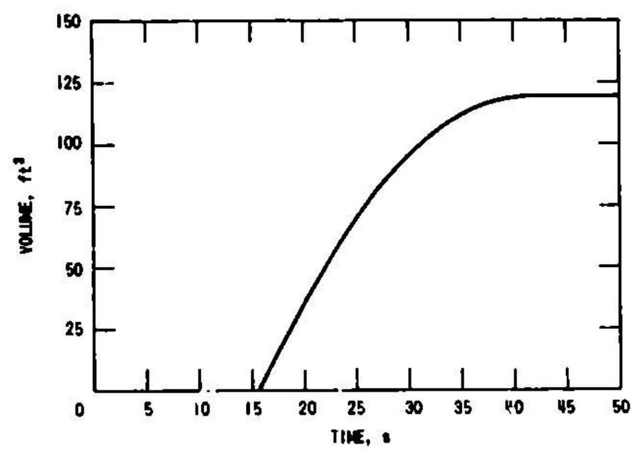


Fig. 55. Sodium Volume, Evaporator Line into Primary Sodium. Conversion factor: 1 ft^3 = 0.028 m^3.

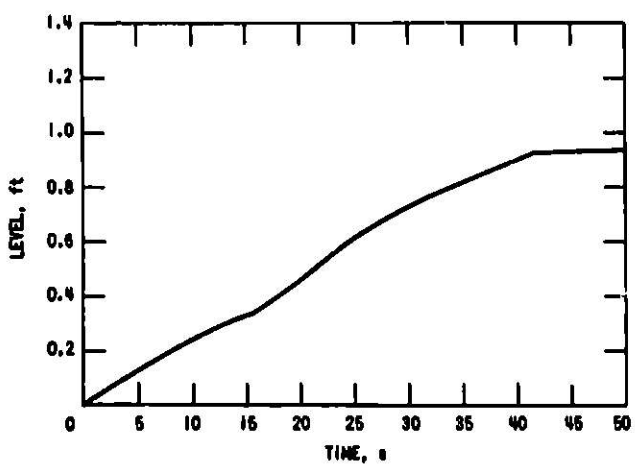


Fig. 56. Sodium-level Change in Primary Tank. Conversion factor: 1 ft = 0.3048 m.

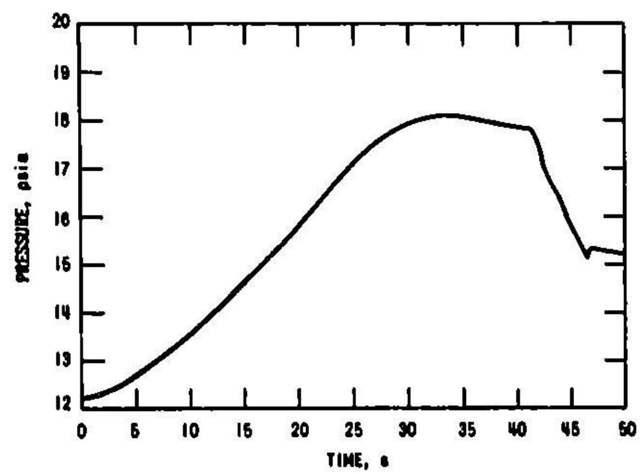


Fig. 57. Argon Pressure in Cover Gas. Conversion factor: 1 psia = 6.895 kPa.

at 17 s. It then alternately increases, decreases, and increases to 3600 gpm (0.23 m^3/s) just before the line is emptied at 41 s. This effect is caused by the interaction of the changing sodium head, the decreasing flow resistance as the line empties, and the changing pressure differential between surge-tank and primary-tank cover gas. The damped oscillations occurring for the next 4.5 s are the result of the pressure of the primary-tank cover gas periodically blowing the small remaining slug of sodium back toward the surge tank as it enters the vertical leg above the IHX.

The height of secondary sodium above the bulk-sodium level decreases as the lines drain. The head is maintained during the first few seconds because of hot-line sodium returning to the surge tank as its inertia decays. The initial 43.6-ft (13.3-m) head decreases to 38 ft (11.6 m) in 8.5 s. At this point the surge-tank outlet to the pump is exposed, and the hot and cold heads are no longer the same. The cold-line head, as shown in Fig. 50, then drops faster for the next 9 s because of a higher flow rate and smaller line.

Figures 52 and 53 show the various vertical, sloping, and horizontal piping runs. The hot-line head, as shown in Fig. 52, clears the surge tank at 17.5 s, reaching the 10-in. (25.4-cm) NA-34-529 and -530 lines at 25 s and the two long 10-in. (25.4-cm) lines beneath the evaporators at 34.5 s. The hot-line sodium reverses flow direction for the second time at 42 s (see Fig. 52). From 34.5 to 42 s, only 9 ft^3 (0.25 m^3) of sodium flows out of the hot line. This marks the point of farthest advance of the argon-sodium interface. It never reaches the first evaporator junction. The remaining sodium in the hot line continues to slosh back and forth near this system low point on a period of about 110 s.

The sodium flow rate in the return line through the superheaters and evaporators (see Fig. 53) decreases smoothly from an initial -5988 gpm (-0.38 m^3/s) to 0 gpm at 7 s to a peak of 2730 gpm (0.17 m^3/s) at 19 s. (For this analysis, positive flow and flow rate are defined as being toward the reactor.) It then decreases to 0 gpm at 42 s. After this second flow reversal, no more sodium from the hot line is discharged into the primary tank.

Figures 54 and 55 show that the sodium volume discharged into the primary tank increases at an average rate of 11 ft^3/s (0.31 m^3/s) to a 465-ft^3 (13.2-m^3) total volume increase after 41.5 s. Another 5 ft^3 (0.14 m^3) is discharged in the next 5 s. No more sodium is dumped into the primary tank after 46.5 s. The rate of discharge increases at 15.5 s, marking the point at which hot-line sodium reaches the primary tank (hot-line flow was back toward the surge tank for 7 s). Figure 56 shows that this amount of sodium raises the primary-tank level 11 in. (28 cm).

Figure 57 indicates a peak of 18.1 psia (124.8 kPa) at 33 s. The pressure then gradually decreases to 17.8 psi (122.7 kPa) at 41.5 s because of a lower rate of compression as the hot-line flow reduces. The pressure then falls rapidly as the hot-line sodium reverses direction and expansion space is provided in the hot line near the IHX. When the cold line finally empties at 46.5 s, the calculation for cold-line flow is switched to the gas-flow subroutine. The 0.2-psi (1.379-kPa) pressure differential between the surge and primary tanks causes an argon flow spike for about 1 s. The argon flow peaks at 720 cfm (0.34 m^3/s). After this peak, the surge- and primary-tank pressures are essentially equal because of the large 12-in. (30-cm) interconnection of the cold line.

APPENDIX C

Analysis of Worst-case Sodium Fire in EBR-II Primary Tank

1. Summary

During the postulated worst-case air-sodium fire in the cover-gas space, the maximum estimated temperature of combustion is 1200°F (649°C). This maximum temperature occurs within 1-2 in. (2.5-5 cm) of the sodium-pool surface. The expected temperature gradient in the cover gas is 200-500°F (111-278°C). The temperature instrumentation would therefore probably not give an alarm for this fire.

Expansion of the cover gas during the worst-case fire is not excessive. Any hypothetical pressurization in the cover-gas space can be nullified by the floating-head tank. In essence, the instrumentation for cover-gas pressure cannot be expected to alarm for the worst-case fire.

The production of sodium oxide during the worst-case fire presents no immediate hazard to reactor operation. Many diverse types of instrumentation serve to detect off-normal conditions. For example, the primary-cover-gas chromatograph would alarm for the worst-case fire within 4 min. Data on primary-sodium plugging temperatures and continuous oxygen-hydrogen meter module readout would be useful in detecting smaller air leaks. Under worst-case hypothetical conditions, sodium burning could occur for 1 h 14 min before the maximum solution capacity of bulk sodium occurs. This time estimate is extremely conservative, but the results indicate no immediate hazard to reactor operation.

2. Definition of Hypothetical Accident Resulting in Worst-case Fire in Primary Tank

The only possible way of introducing significant quantities of air into the primary cover gas is by rupturing the suction side of the FUM (fuel-unloading machine) turbine while blowing argon over a fueled subassembly and through the primary tank. The FUM turbine is rated for 150 cfm (0.07 m^3/s) at 5 psi (34.5 kPa); however, the nominal output is 60-70 cfm (~0.03 m^3/s) while the air is blowing past a subassembly. This flow rate could be slightly higher as a result of reduced pressure loss if the inlet piping from the A-3 nozzle were bypassed. For the purposes of analysis, a maximum rate of [illegible] cfm (0.035 m^3/s) will be assumed. At this rate, the oxygen density is

$$\rho_{O_2} = 0.07528\ \text{lb/ft}^3 \times 0.232 \times \frac{12.3\ \text{psi (EBR-II)}}{14.7\ \text{psi (std)}}$$

$$= 0.01461\ \text{lb}\ O_2/\text{ft}^3\ \text{air}\ (0.0234\ \text{kg}\ O_2/\text{m}^3\ \text{air}),$$

where

0.232 = ratio of oxygen to air volume

and

0.07528 = density of oxygen.

The mass flow rate of oxygen is

$$W_{O_2} = 75\ ft^3/min \times 0.01461\ lb/ft^3 = 1.096\ lb/min\ (0.0083\ kg/s).$$

For this calculation, standard temperature is assumed. The effect of heating by the inlet heaters would be to reduce the mass flow rate. If all the oxygen is assumed converted to sodium oxide, the combustion will require 3.15 lb/min (0.024 kg/s) of sodium and will produce 4.25 lb/min (0.03 kg/s) of sodium oxide.

2.1 General Observations on Pool-type Sodium Fires in Air

a. Although various reported experiments have yielded sodium combustion rates of 0.1-0.3 $lb/ft^2 \cdot min$ (0.01-0.02 $kg/m^2 \cdot s$), the most reliable indicates a range of 0.12-0.167 $lb/ft^2 \cdot min$ (0.010-0.013 $kg/m^2 \cdot s$), or 5-10 $lb/ft^2 \cdot h$ · (35.3-49.0 $kg/m^2 \cdot h$).[23] The combustion rate increases with increasing surface area and depth

b. The sodium-pool temperature increases gradually. The rate is a function of the particular system mass, heat transfer, material, geometry, etc. The maximum possible temperature is 1616°F (880°C), the boiling point of sodium.[19,23]

c. A high-temperature gradient occurs vertically from the pool surface. Reference 24 indicates gradients of 500°F/in. (10.9°C/mm) as typical. Data from Ref. 19 show gradients of 200°F/in. (4.37°C/mm).

d. Predictions as to the behavior of the sodium oxide are varied. Reference 19 indicates that most of the oxide may settle to the pool surface uniformly. This settling will be subject to drafts because the oxide settles as a fine powder. Reference 23 states (on p. 23), "a sodium oxide aerosol will deposit uniformly over the inside surfaces of the enclosure ... ."

2.2 General Observations on Worst-case Fire

a. If the surface of primary bulk sodium were completely open to the atmosphere, at an assumed burning rate of 0.167 lb/ft^2 (0.82 kg/m^2) and a pool surface of 531 ft^2 (49 m^2) the sodium would be consumed at 88.7 lb/min (0.67 kg/s). However, because the maximum supply of air is 75 cfm (0.04 m^3/s), the previously calculated value of 3.15 lb/min (0.024 kg/s) of sodium is the limiting value.

b. The average temperature of the bulk sodium will not increase during the worst-case fire. The secondary sodium system would compensate for this very small and slow addition of heat. A small temperature gradient of about 30°F/in. (0.67°C/mm) will exist near the pool surface.[24]

c. The maximum temperature of the combustion is subject to speculation. The range is from less than 1000°F (538°C) to a theoretical maximum of 5000°F (2760°C). Maximum experimental combustion temperatures of 3600°F (1982°C) have been obtained with optimum ratios and injection techniques. Temperatures associated with sodium-pool reactions are much lower than the maxima. Reference 19 indicates that temperatures of 900-1500°F (482-816°C) occur close to the pool surface.

d. The temperature gradients measured in experiments indicate that the cover-gas temperature should be near ambient (625°F, or 329°C) a few inches above the pool surface. The values quoted in Refs. 19 and 24 indicate that the primary-tank cover would receive very little heat, and irrespective of alarm setting, it is doubtful that the thermocouples in the gas space would sense any temperature increase resulting from the worst-case fire or any lesser fire.

2.3 Sodium Oxide Buildup

If we use the assumption of uniform oxide deposition from the calculated surface area of Ref. 23, the cover-gas space [neglecting all surface areas other than the flat surfaces of the tank cover, 15 in. (38 cm) of side wall, and the pool surface] is 1164 ft^2 (108 m^2). At an oxide production rate of 4.25 lb/min (0.03 kg/s), the buildup will proceed at 0.00365 $lb/ft^2 \cdot min$ (0.0003 $kg/m^2 \cdot s$). Of this, 2.31 lb/min (0.02 kg/s) will deposit on the surface of the inner-tank cover and on the side wall, and 1.94 lb/min (0.015 kg/s) will settle to the pool surface to eventually dissolve.

The buildup rate on the wall and top cover, calculated by using a density of 142 lb/ft^3 (2.27 g/cm^3), is 0.0102 in./h (0.0259 cm/h)--a very slow rate.

No information is available to indicate the rate at which the oxide would dissolve in the bulk sodium. If it were to dissolve at the buildup rate (assuming uniform deposition), the oxygen concentration would increase at 0.77 ppm/min. The maximum plugging temperature of 325°F (163°C) would be attained in about 3 min. If all the oxide produced settled onto the pool and dissolved, the oxygen concentration would increase at 1.72 ppm/min, and the maximum plugging temperature would be attained in 1.16 min. These times are based on the total oxide produced dissolving as it is created. The 325°F (163°C) plugging temperature is an operating limit, however, and is considerably below the limit at which plugging could start in the reactor.

At 580°F (304°C),* the plugging concentration (or maximum solubility) is 126.5 ppm.** At an assumed solution rate of 1.72 ppm/min, the bulk sodium would reach a maximum solubility in 1 h 14 min. At the normal bulk-sodium operating temperature of 700°F (371°C), the maximum solubility is 403 ppm. At normal bulk-sodium operating temperature, this solubility would not be reached for 3 h 54 min.

At 580°F (304°C), the reactor would be shut down, in fuel-handling operations, starting operation, or at 5% or less power. During this type of operation, 1 h 14 min is more than adequate time to detect the condition and take corrective action.

2.4 Predicted Temperature and Pressure Response

For this analysis, assume that two burning rates may exist:

a. An initial rate based on the amount of sodium vapor existing in saturation at the ambient cover-gas temperature-pressure conditions.

b. A steady-state rate based on sodium-pool-burning information.

2.4.1 Initial Burning Rate

Theoretically, the concentration in argon cover gas of sodium vapor and aerosol at power is 140 ppm. However, other factors (oxides and impurities acting as deposition sites) may raise the concentration by a factor of 7-10. A value of 1400 ppm is considered an upper limit.

The value for argon density at power is necessary to calculate the heat rise due to the initial burning rate. The indicated cover-gas temperature over many runs is 650 ± 10°F (343 ± 5.6°C).

The formula for density is

$$\rho = \frac{P}{ZRT},$$

where

Z = proportionality constant = 1.0,

R = 1546/m = 38.5,

T = 650 + 460 = 1110°F (599°C),

*The minimum operating temperature for primary bulk sodium is 580°F (304°C). Maximum power at this temperature is 5% of full power.

**$\log_{10} O_2$ (in ppm) = 6.967 - 2809/K.

and

P = pressure = 12.3 psi (84.8 kPa).

The argon density therefore is

$$\rho = \frac{12.3 \times 144}{1 \times 38.5 \times 1110} = 0.04145 \text{ lb/ft}^3 \ (0.0663 \text{ kg/m}^3).$$

Volume = V = hA = $(15/12) \times 531$ = 663.75 ft^3 (18.8 m^3).

Mass of argon* = 0.04145×663.75 = 27.51 lb (12.5 kg).

Mass of sodium in argon** = $27.51 \times 1400 \times 10^{-6}$

= 0.03851 lb (0.0175 kg).

If 4500 Btu/lb of sodium (4.7 mJ/lb) is assumed for complete combustion,[23] the heat released by 0.0385 lb (0.0175 kg) of sodium is

$$Q = 4500 \times 0.0385 = 173.3 \text{ Btu } (182.8 \text{ kJ}).$$

The total heat rise in the argon, calculated by using a specific heat of 0.124, is

$$\Delta T = \frac{173.3}{27.51 \times 0.124} = 51°\text{F } (28°\text{C}).$$

This calculation is based on complete mixing. The value does not account for losses due to heat transfer between the heated gases and the bulk sodium and the vessel. The maximum temperature rise would also be limited by the oxygen injection rate (incomplete mixing) and by the venting of the heat from the vessel as the FUM turbine continues to supply air.

With an assumed 51°F (28°C) temperature rise (and the further assumption that the pressure-control systems are inoperative), the pressure associated with this increase is

$$P = 12.3 \times \frac{1161}{1110} = 12.86 \text{ psi } (88.67 \text{ kPa}).$$

This is an increase of 0.56 psi (3.9 kPa) or 15.7 in. of water, well within the capability of the control systems for reactor pressure (floating-head tank and the pressure-vacuum relief system). The actual pressure rise due to sodium-vapor combustion would be much lower than this theoretical figure and would also be regulated by the relief-system characteristics when driven by the blower flow under the assumed conditions.

*Mass of argon in primary tank at 12.3 psi (84.8 kPa) and 650°F (343°C).

**At concentration of 1400 ppm.

2.4.2 Steady-state Burning Rate

The large vertical temperature gradient experimentally assumed to exist above a burning sodium pool requires that most of the heat of combustion [4500 Btu/lb (4.7 mJ/lb)] be absorbed by the pool. Experimental data[19] indicate that the pressure increase due to the heat of combustion is slight. The reference indicates that the maximum pressure differential was 0.46 in. H_2O (114.5 Pa) with the following initial conditions:

Air leakage rate: 200 cfm (0.094 m^3/s) at a pressure differential of 0.1 in. H_2O (24.9 Pa).

Burning area: 32 ft^2 (2.98 m^2).

Sodium mass: 700 lb (317.5 kg).

Sodium temperature: 1080°F (582°C).

Test-chamber volume: 3350 ft^3 (94.8 m^3). (This volume is about five times as great as the cover-gas space.)

This demonstrates that much of the heat is absorbed by the sodium pool.

For comparison, assume that a layer of hot gas extends 6 in. (15 cm) above the pool at 1200°F (649°C). An effective burning area computed according to the air-injection rate of 3.15 lb/min (0.024 kg/s) (see Sec. 2.1 of this appendix) at a combustion rate of 0.167 lb/ft^2·min (0.0136 kg/m^2·s) is

$$A_{eff} = 3.15/0.167 = 18.86 \text{ ft}^2 \text{ } (1.75 \text{ m}^2).$$

The combustion volume, assuming 6 in. (15 cm) of hot gas above the pool, is 9.43 ft^3 (0.27 m^3). For constant volume (no pressure regulation), the increased pressure at a combustion temperature of 1200°F (649°C) is

$$P_{V_{eff}} = 12.3 \times \frac{1660}{1110} = 18.39 \text{ psi (126.8 kPa)}.$$

The change in pressure is

$$\Delta P = 18.39 - 12.3 = 6.09 \text{ psi (41.99 kPa)}.$$

For the total volume of the cover-gas space, this increase amounts to 0.08 psi or 2.39 in. H_2O (~575 Pa). Allowing for the differences in volumetric ratio of 5, the results agree favorably with the experimental data of Ref. 19. This value, added to the 1-in. (2.5-cm) differential already in the primary cover gas, would give a cover-gas differential of 3.34 in. (8.5 cm) for the worst-case fire. That is, without any pressure regulation, the cover-gas pressure would not reach the normal trip point of 3.5 in. H_2O (871 Pa). [This pressure rise should, however, activate the high-pressure alarms at the outlets of the N-1 and -2 nozzles, which are set at 2 ± 0.25 in. H_2O (498 ± 62 Pa).]

However, the floating-head tank will nullify any pressure changes within the primary-cover-gas space. If the pressure-regulation system is assumed to function (if the floating-head tank has been very reliable), the volume of expansion will be

$$V_{ep} = 9.43 \times \frac{1660}{1110} = 14.1 \text{ ft}^3 \ (0.4 \text{ m}^3).$$

This value gives a volumetric change of

$$V = 14.1 - 9.43 = 4.67 \text{ ft}^3 \ (0.13 \text{ m}^3).$$

In summary, the floating-head tank will compensate for the volumetric expansion, even at its upper travel limit, which has a safety factor of 5 ft^3 (0.14 m^3). Analysis indicates that the tank could respond to this change in 45 ms. If the tank could not function, the pressure rise due to the worst-case fire would still have a probability of not being detected; but in any case, the pressure rise would not cause the pressure-relief system to actuate, nor would the oil seals of the floating-head tank be blown. Lesser fires would be virtually impossible to detect.

3. Conclusions

Under the conditions of the postulated worst-case fire, (a) a trip for temperature of primary cover gas set at 710°F (377°C) would not actuate, (b) a pressure trip set at 3.5 in. H_2O (871 Pa) would not actuate, and (c) no immediate hazard to reactor operation would exist.

If the reactor were at its worst possible (with respect to the fire) operating state with 5% power and a bulk-storage-tank temperature of 580°F (304°C), the fire could continue for more than 1 h 14 min before the solubility limit of sodium oxide would be reached. The primary-gas chromatograph would alarm this condition within 4 min. (The data are updated every 4 min; the nitrogen concentration would exceed its 7000-ppm limit within 4 s.) Smaller leaks would be detected by the primary-sodium plugging temperature and the continuous oxygen-hydrogen meter module. Thus, existence of a 75-cfm (0.035-m^3/s) air-sodium combustion for an hour without detection is inconceivable. If all else fails at power, the PPS instrumentation for primary flow would detect the reduction in flow as plugging of the inlet orifices started, and the instrumentation would shut down the reactor before the flow had decreased much below the flow trip point.

APPENDIX D

Approximate Stress Analysis of Bottom of Inner Vessel of EBR-II Primary Tank

The bottom structure of the inner vessel is designed to support a load of 500 tons (453 592 kg). The bottom consists of a 1.5-in (3.8-cm)-thick plate and knuckle piece, reinforced with eight parallel beams about 21 in. (53.3 cm) deep. Running across the top of these eight beams and forming a common top flange for them is a 0.75-in. (1.9-cm)-thick convection plate about 12.5 ft (3.81 cm) wide with 4-in. (10.16-cm)-dia holes randomly located in it to facilitate the convection flow of the sodium. Plug-welded to the top of the convection plate is a circular leveling plate, 1.5 in. (3.8 cm) thick and 11 ft 10 in. (3.6 m) in diameter. Shallower beams and ribs perpendicular to the main set of eight beams serve mainly to stiffen the bottom plate.

For the analysis, the inner-vessel bottom is simplified by considering a simply supported square plate stiffened with stiffeners and ribs running perpendicularly across the plate. The stiffeners and ribs are assumed to be averaged over the stiffener and rib spaces. The plate is considered to be loaded by a load distributed uniformly over the bottom of the reactor vessel.

1. Theoretical Derivations

The theory was derived by specifying strain energy of the plate corresponding to the linear theory of elasticity and by applying the method of minimum potential energy to obtain equilibrium equations and natural boundary conditions. If the following symbols are defined by

a = plate length,

D_{ij} = loading stiffness of the plate,

d = stiffener spacing,

E, E_S, E_R = Young's moduli of the plate, the stiffener, and the rib, respectively,

G, G_S, G_R = shear moduli of the plate, the stiffener, and the rib, respectively,

h = thickness of the plate,

I_S, I_R = moments of inertia of the stiffener and rib about the midplane of the plate,

J_S, J_R = torsional constant of the stiffener and rib,

ℓ = rib spacing,

M_x, M_y = moment resultants of the plate,

$q(x, y)$ = applied lateral load,

w = displacement of the plate in z direction,

x, y = coordinates of the plate (see Fig. 58),

ν = Poisson's ratio,

and

σ_x, σ_y = stresses of the plate,

the equation of equilibrium can be written as

$$D_{11}\frac{\partial^4 w}{\partial x^4} + 2(D_{66} + D_{12})\frac{\partial^4 w}{\partial x^2 \partial y^2} + D_{22}\frac{\partial^4 w}{\partial y^4} = q(x, y), \tag{1}$$

where the values of D_{ij} are defined by[25]

$$\left.\begin{aligned}
D_{11} &= \frac{Eh^3}{12(1 - \nu^2)} + \frac{E_S I_S}{d}, \\
D_{22} &= \frac{Eh^3}{12(1 - \nu^2)} + \frac{E_R I_R}{\ell}, \\
D_{12} &= \frac{\nu E h^3}{12(1 - \nu^2)}, \\
&\text{and} \\
D_{66} &= \frac{Eh^3}{24(1 + \nu)} + \frac{G_S J_S}{2d} + \frac{G_R J_R}{2\ell}.
\end{aligned}\right\} \tag{2}$$

When a square plate is simply supported, the boundary conditions can be written as

$$\left.\begin{aligned}
x &= 0,\ a,\quad w = 0, \\
x &= 0,\ a,\quad \frac{\partial^2 w}{\partial x^2} = 0;
\end{aligned}\right\} \tag{3}$$

and

$$\left.\begin{aligned}
y &= 0,\ a,\quad w = 0, \\
y &= 0,\ a,\quad \frac{\partial^2 w}{\partial y^2}.
\end{aligned}\right\} \tag{4}$$

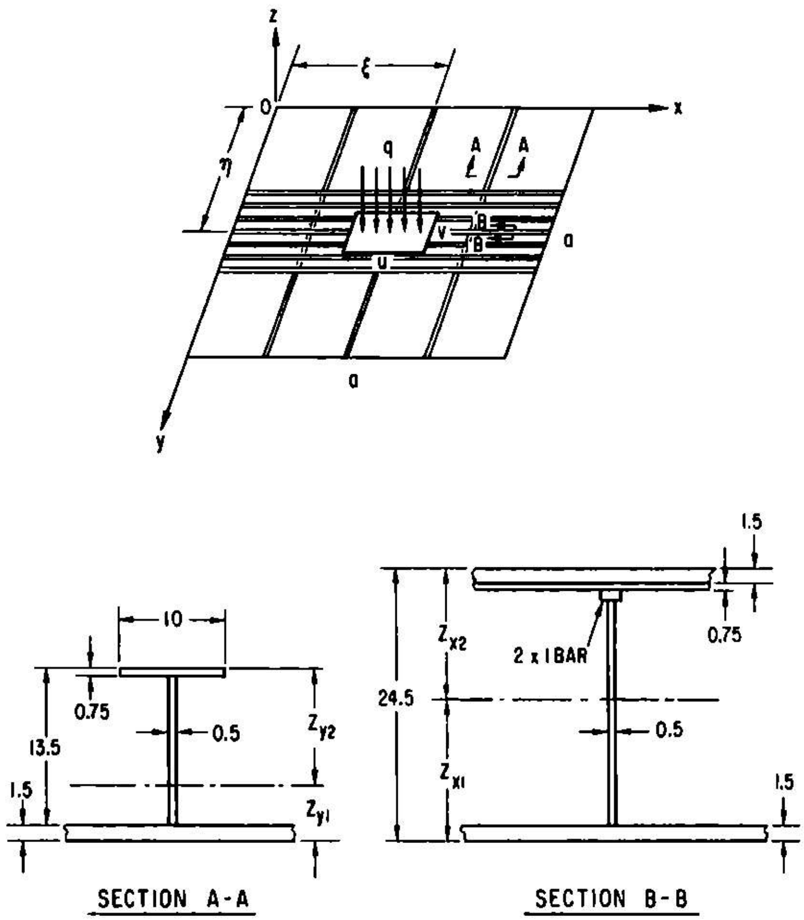


Fig. 58. Geometry of Plate for Stress Analysis of Bottom Structure of Inner Vessel of Primary Tank. (All dimensions in inches; 1 in. = 25.4 mm.)

The Navier Solution of Eq. 1 for a simply supported plate can be obtained similarly to that of Ref. 26:

$$w = \frac{16qa^4}{\pi^6} \sum_{m=1,3,\ldots}^{\infty} \sum_{n=1,3,\ldots}^{\infty} \frac{\sin\frac{m\pi x}{a} \sin\frac{n\pi y}{a} \sin\frac{m\pi \xi}{a} \sin\frac{n\pi \eta}{a} \sin\frac{m\pi u}{2a} \sin\frac{n\pi v}{2a}}{mn[m^4 D_{11} + 2m^2 n^2 (D_{66} + D_{12}) + n^4 D_{22}]}, \tag{5}$$

where the pressure loading q is distributed over area uv in Fig. 58, and ξ and η are the distances from the boundary to the load center as shown in that figure.

The moment resultants in x and y directions are defined by

$$\left.\begin{aligned} M_x &= -\left(D_{11}\frac{\partial^2 w}{\partial x^2} + D_{12}\frac{\partial^2 w}{\partial y^2}\right) \\ &\text{and} \\ M_y &= -\left(D_{22}\frac{\partial^2 w}{\partial y^2} + D_{12}\frac{\partial^2 w}{\partial x^2}\right) \end{aligned}\right\}. \tag{6}$$

Substituting Eq. 5 into Eq. 6 gives

$$\left.\begin{aligned}
M_x &= \frac{16qa^2}{\pi^4}\sum_{m=1,3,\ldots}^{\infty}\sum_{n=1,3,\ldots}^{\infty}\frac{(m^2D_{11}+n^2D_{12})\sin\frac{m\pi x}{a}\sin\frac{n\pi y}{a}\sin\frac{m\pi\xi}{a}\sin\frac{n\pi\eta}{a}\sin\frac{m\pi u}{2a}\sin\frac{n\pi v}{2a}}{mn[m^4D_{11}+2m^2n^2(D_{66}+D_{12})+n^4D_{22}]} \\
&\text{and} \\
M_y &= \frac{16qa^2}{\pi^4}\sum_{m=1,3,\ldots}^{\infty}\sum_{n=1,3,\ldots}^{\infty}\frac{(m^2D_{12}+n^2D_{22})\sin\frac{m\pi x}{a}\sin\frac{n\pi y}{a}\sin\frac{m\pi\xi}{a}\sin\frac{n\pi\eta}{a}\sin\frac{m\pi u}{2a}\sin\frac{n\pi v}{2a}}{mn[m^4D_{11}+2m^2n^2(D_{66}+D_{12})+n^4D_{22}]}
\end{aligned}\right\}. \tag{7}$$

Equation 7 is a rapidly converging series, so a satisfactory approximation can be obtained by taking a few terms of the series. If only one term of the series is taken and the load center is at the center of the plate, then the maximum moment resultants, which occur at the center of the plate, can be found as

$$\left.\begin{aligned}
(M_x)_{max} &= \frac{16a^2(D_{11}+D_{12})\sin\frac{m\pi u}{2a}\sin\frac{n\pi v}{2a}}{D_{11}+2(D_{66}+D_{12})+D_{22}} \\
&\text{and} \\
(M_y)_{max} &= \frac{16a^2(D_{12}+D_{22})\sin\frac{m\pi u}{2a}\sin\frac{n\pi v}{2a}}{D_{11}+2(D_{66}+D_{12})+D_{22}}
\end{aligned}\right\}. \tag{8}$$

The stresses of the plate can be found as

$$\left.\begin{aligned}
\sigma_x &= \frac{EZ_x}{(1-\nu^2)D_{11}}M_x \\
&\text{and} \\
\sigma_y &= \frac{EZ_y}{(1-\nu^2)D_{22}}M_y
\end{aligned}\right\}. \tag{9}$$

where Z_x and Z_y are the distances along the z axis from neutral (zero-stress) surfaces of the plate to the points being considered.

2. Numerical Calculations

The numerical values of the plate are

$$
\left.
\begin{aligned}
E &= E_S = E_R \quad 24.5 \times 10^6 \text{ psi } (168.9 \text{ GPa}), \\
\nu &= 0.3, \\
G &= G_S = G_R = 9.42 \times 10^6 \text{ psi } (64.9 \text{ GPa}), \\
a &= 276 \text{ in. } (7.02 \text{ m}), \\
d &= 19 \text{ in. } (48 \text{ cm}), \\
\ell &= 65 \text{ in. } (1.65 \text{ m}), \\
q &= 153 \text{ psi } (1.055 \text{ kPa}), \\
u &= 80 \text{ in. } (2.03 \text{ m}),
\end{aligned}
\right\} \tag{10}
$$

and

$$v \quad 80 \text{ in. } (2.03 \text{ m}).$$

The following values can be obtained:

$$
\begin{aligned}
I_S \quad & \tfrac{1}{3} \times 2 \times (1)^3 + (20.5)^2 \times (1 \times 2) + \tfrac{1}{3} \times 0.5 \times (19.75)^3 + (0.75)^2 \\
& \times (0.5 \times 19.75) + \left[\frac{(138)^2 \times (1.5)^3}{3 \times (312)^2} + \frac{(138)^2 \times 1.5 \times (22.25)^2}{(312)^2} \right. \\
& \left. + \frac{260 \times 150 \times 4 \times (0.75)^4}{3 \times \pi \times (312)^2 \times 1.5} + \frac{260 \times 150 \times 4 \times (0.75)2 \times (21.5)^2}{\pi \times (312)^2 \times 1.5} \right] \times 19 \\
& \approx 6575 \text{ in.}^4 \ (273\,672 \text{ cm}^4), \\
I_R \quad & \tfrac{1}{3}[10 \times (0.75)^3 + 0.5 \times (12.75)^3] + 10 \times 0.75 \times (13.5)^2 + 0.5 \times 12.75 \\
& \times (0.75)^2 \quad 1717.3 \text{ in.}^4 \ (71\,479 \text{ cm}^4), \\
J_S \quad & 0.229 \times 2 \times 1 + \tfrac{1}{3} \times 19.75 \times (0.5)^3 + \tfrac{1}{3} \times 19 \times (1.5 \times 0.1596)^3 + \tfrac{1}{3} \\
& \times 19 \times (0.75 \times 0.255)^3 \quad 1.41 \text{ in.}^4 \ (58.7 \text{ cm}^4), \\
J_R \quad & \tfrac{1}{3}[10 \times (0.75)^3 + 12.75 \times (0.5)^3] \quad 1.94 \text{ in.}^4 \ (80.74 \text{ cm}^4), \\
Z_{x1} \quad & \frac{19 \int_0^{1.5} Z\,dZ + 0.5 \int_{1.5}^{19.75} Z\,dZ + 2 \int_{19.75}^{20.75} Z\,dZ + 19 \int_{20.75}^{22.25} Z\,dZ}{19 \times 1.5 + 19.75 \times 0.5 + 2 \times 1 + 2.25 \times 19} \\
& 13.9 \text{ in. } (35.3 \text{ cm}),
\end{aligned}
$$

and

$$
Z_{y1} \quad \frac{65 \int_0^{1.5} Z\,dZ + \int_{1.5}^{12.75} Z\,dZ + \int_{12.75}^{13.75}}{65 \times 15 + 0.5 \times 12.75 + 10 \times 0.5} \quad 2.9 \text{ in. } (7.366 \text{ cm}). \tag{11}
$$

where Z_{x1} and Z_{y1} are the distances from the neutral axes of the stiffener and rib to the bottom surface of the plate, as shown in Fig. 58

Substituting Eqs. 10 and 11 into Eq. 2 yields

$$\left.\begin{aligned}
D_{11} &= 8486.3 \times 10^6 \text{ lb·in.} \quad (958.8 \times 10^6 \text{ N·m}),\\
D_{22} &= 650.8 \times 10^6 \text{ lb·in.} \quad (73.44 \times 10^6 \text{ N·m}),\\
D_{12} &= 7.57 \times 10^6 \text{ lb·in.} \quad (8.55 \times 10^6 \text{ N·m}),\\
\text{and}\\
D_{66} &= 2.66 \times 10^6 \text{ lb·in.} \quad (3 \times 10^6 \text{ N·m}).
\end{aligned}\right\} \tag{12}$$

The maximum deformation and bending stresses of the bottom structure can be found from Eqs. 5, 8, and 12:

$$\left.\begin{aligned}
w_{max} &= 0.3 \text{ in. } (0.784 \text{ cm}),\\
(\sigma_x)_{max} &= 10\,574 \text{ psi } (72\,908 \text{ kPa}),\\
\text{and}\\
(\sigma_y)_{max} &= 9\,205 \text{ psi } (63\,486 \text{ kPa})
\end{aligned}\right\} \tag{13}$$

The stress in the plate is found to be less than the design limit of $1.5S_m$, where S_m is the allowable stress intensity. For Type 304 stainless steel at 700°F (371°C), the value of S_m is 15 600 psi (107 559 kPa).

3. Discussion

The above results were based on several assumptions. First, the circular bottom structure is simulated by a simply supported square plate, the dimensions of which are based on the equivalent area of the circular bottom plate. It is conservative to assume that the plate is simply supported, since the maximum stress obtained by this boundary condition gives higher maximum stress than that obtained by using the actual boundary condition of the bottom structure. Second, the leveling and convection plates are assumed to be homogenized over the stiffeners, and in addition, the stiffeners and ribs are assumed to be homogenized over the space between them. The homogenizing is feasible, since the stiffeners are closely spaced. The effect of ribs probably should be individually counted for through introduction of the Dirac delta or impulse function as described in Refs. 27 and 28. However, the numerical calculation of a plate with discrete rib properties is not only lengthy, but also cumbersome.

The results obtained from the analysis are conservative. In reality, the sodium weight of 320 tons (290 000 kg) is uniformly distributed over the bottom structure. In the analysis, however, the sodium weight was assumed to be applied only to the area uv, as shown in Fig. 58.

APPENDIX E

Effects of Stresses in Primary Tank and Associated Structures When Temperature of Primary-tank Sodium Increases from 700 to 800°F (371 to 427°C)

The stress analysis of the EBR-II primary tank and its associated structures presented in Ref. 29 was based on the assumption that the tank sodium temperature was 700°F (371°C); the temperature gradient, ΔT, across the tank cover was predicted to be 175-190°F (97-106°C) when the heater in the cover is not in operation. It is conservative to assume that, when the primary-sodium temperature increases from 700 to 800°F (371 to 427°C), the ΔT across the primary-tank cover increases from 190 to 290°F (106 to 143°C).

The inner-vessel wall has a membrane stress of 2100 psi (14.5 MPa) due to mechanical dead load and a maximum thermal bending stress of 8200 psi (56.5 MPa) due to a ΔT of 175°F (97°C) across the primary-tank cover.[29] At a ΔT of 290°F (143°C), the maximum thermal bending stress is 13 590 psi (93.7 MPa) and the maximum combined mechanical and thermal stress is 15 690 psi (108.2 MPa).

The outer-vessel wall is subject to dead loads that produce a tensile stress of 2800 psi (19.3 MPa) and to a ΔT of 175°F (97°C) across the primary cover that produces a maximum thermal-bending stress of 5700 psi (39.3 MPa).[29] At a ΔT of 290°F (161°C), the maximum thermal-bending stress is 9450 psi (65.2 MPa) and the maximum combined mechanical and thermal stress is 12 250 psi (184.5 MPa).

The maximum thermal bending stress in the radial beam of the primary tank, $(f_y)_{max}$, can be expressed as[29]

$$(f_y)_{max} = 36.5\Delta T,$$

and the maximum bending stress due to mechanical loading is 4830 psi (33.3 MPa). When ΔT across the cover is assumed to be 290°F (161°C), $(f_y)_{max}$ is 10 590 psi (73.0 MPa) and the maximum combined bending stress is 15 420 psi (106.3 MPa).

Based on Section III of the ASME code,[18] the calculated primary plus secondary* stress itensity should be equal to or less than $3S_m$ (S_m is the allowable stress intensity). For Type 304 stainless steel at 800°F (427°C), S_m equals 14 400 psi (99.3 MPa). Thus, the combined mechanical and thermal stresses in the inner-vessel wall, in the outer-vessel wall, and in the radial beam of the primary tank are less than $3S_m$ [i.e., 43 200 psi (297.9 MPa)], and they are permitted by the ASME code.

*Thermal stress is considered to be a secondary stress.

APPENDIX F

Analysis of Thermal Response of EBR-II Reactor Core during Secondary Malfunction of System

1. Summary

Two general modes of secondary-system malfunction are possible. These modes are categorized not in terms of specific hardware failure, but with respect to their effect upon the primary system and the reactor. The malfunctions are (a) the secondary system overcooling the primary bulk sodium and (b) loss of secondary cooling capacity.

For overcooling, an extreme hypothetical bulk-sodium (reactor-inlet) cooling rate of -0.6°F/s (-0.33°C/s) was selected as an analytical driving function. With no operator action to maintain a 183°F (102°C) core ΔT, a maximum increase in fuel-surface temperature of about 20°F (11°C) occurred in the hottest element in row 6. The transient was terminated by a worst-case trip on 115% of full power 65 s after initiation. At this time, the bulk-sodium temperature was 661°F (349°C). The cooling rate of -0.6°F/s (-0.33°C/s) was selected for analysis to represent about twice the normal cooling capacity at 62.5 MWt and much above the capacity of the secondary system to remove heat. If, during an overcooling incident, the operator noted the increasing power and withdrew control rods to maintain normal reactor power, the subassembly-outlet and fuel temperatures would decrease as power was held constant.

An accident of loss of secondary cooling was analyzed using a temperature rate ($\dot{T}$) of 0.6°F/s (0.33°C/s) for heating of the bulk sodium. No operator intervention was assumed. The results indicate no appreciable temperature rise in any of the fueled subassemblies. For the analysis, the transient assumed was ended after 80 s; at this time, the fuel temperatures were slightly less than their nominal values. For example, the fuel-surface temperature of the hottest element in row 6 had decreased by about 20°F (11°C). This trend would continue until the core became subcritical. At this time, fuel and coolant temperatures would stabilize. For example, the core configuration used in this analysis is at zero power and critical at a bulk-sodium temperature of about 777°F (414°C).

A loss of secondary cooling was also analyzed with the assumption that the operator maintained normal reactor power. The temperature rate of 0.6°F/s (0.33°C/s) was also used in this analysis. Under the assumed conditions, a reactor shutdown due to high subassembly outlet temperature would occur in about 50 s at a bulk-sodium temperature of about 730°F (388°C) and a maximum fuel-surface temperature of about 1105°F (596°C).

2. Discussion

The EROS code[20] was used to obtain the results of this study. Nuclear input data used in the analysis were given in Table XIV. These data are for core configuration 56''' (essentially the core loading for run 56C), which was chosen as a well-characterized core representative of operation with the stainless steel reflector. Major items of importance to ensure applicability of such a core as a base case for kinetic simulation of temperature excursions in the secondary cooling system are that the power and flow distributions, nuclear parameters, and reactivity-feedback characteristics be representative of current and planned core loadings.

2.1 Overcooling Analysis

Figure 59 is a plot of normalized (at 62.5 MWt) power versus temperature during the overcooling incident. The shutdown due to 1.5% power level occurs at 65 s. Figures 60-62 are plots of the response of fuel and coolant temperatures to the overcooled bulk sodium. The curve labeled "inner-surface cladding" is the cladding temperature 1 mil (25.4 μm) from the inner surface.

Figure 61--a plot of change in temperature of the hottest element in row 6, the hottest row in the core--indicates a slight drop in temperature during the first 15 s. This drop is followed by a temperature rise until shutdown initiation at 65 s. The highest fuel-surface temperature is about 1085°F (585°C). Applying an uncertainty factor of 1.157 to the gross temperature increase from the initial bulk-sodium temperature shows that the fuel-surface temperature could reach a maximum of 1145°F (618°C) briefly.

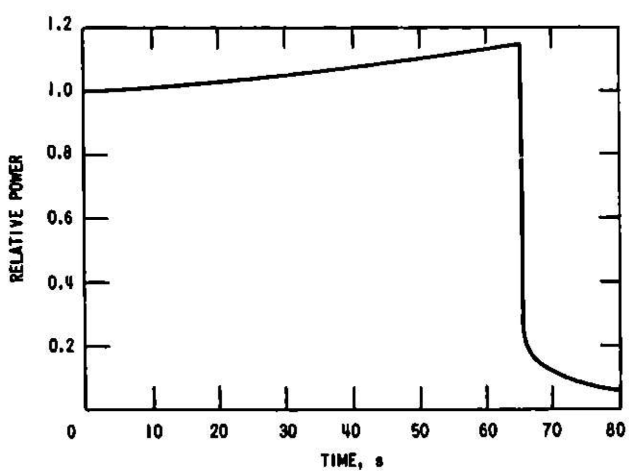


Fig. 59. Power History for Analysis of Overcooling of Primary Sodium

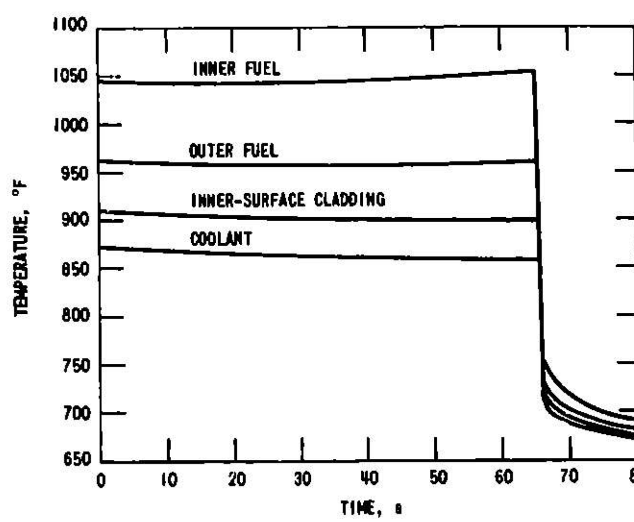


Fig. 60. Temperature of Average Driver-fuel Element in Row 2: Overcooling Analysis. Conversion factor: °C = (°F - 32)/1.8.

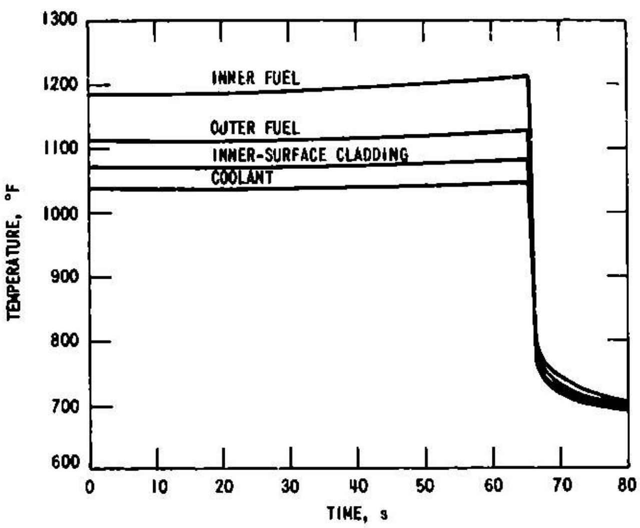


Fig. 61. Temperature of Hottest Driver-fuel Element in Row 6: Overcooling Analysis. Conversion factor: °C = (°F − 32)/1.8.

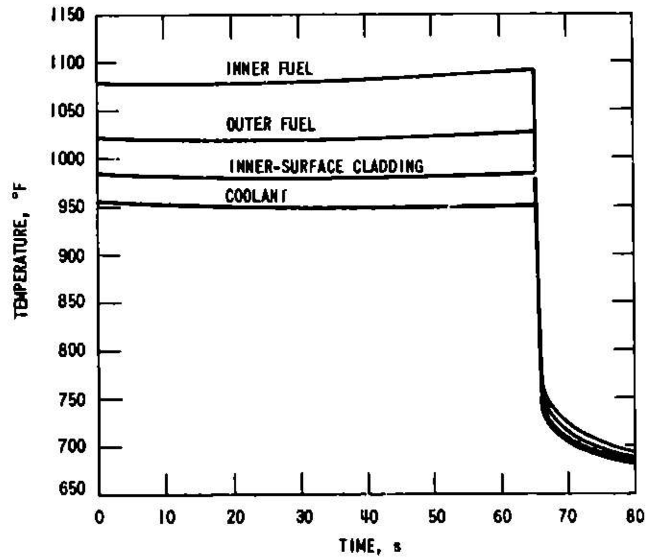


Fig. 62. Temperature of Average Driver-fuel Element in Row 6: Overcooling Analysis. Conversion factor: °C = (°F − 32)/1.8.

The safety limits developed to ensure that loss of cladding integrity is of low probability and is not a safety hazard are as follows:

a. The temperature for formation of a fuel-cladding eutectic shall not be exceeded for more than 60 s.

b. Temperatures at the fuel-cladding interface shall not exceed 1500°F (816°C).

The temperatures of eutectic formation for the Mark-IA and the Mark-II fuel assemblies are 1290 and 1319°F (699 and 715°C), respectively. The results indicate that this excursion is well within the safety limits.

At 65 s, when the trip occurs, the inlet temperature of bulk sodium has dropped 39°F (22°C) to 661°F (349°C) and power has increased to 115% of full power. The coolant temperature differential for an average fuel element in row 2, as calculated from Fig. 59, increased from 170 to 197°F (94 to 109°C).

During an overcooling incident, the operator would be confronted with increasing nuclear power and increasing differential in core coolant temperature. If the condition causing the power and temperature excursions is not immediately detected, the operator would respond by decreasing nuclear power. The reactor would not be endangered by this response, because the overall effect would be a decrease in core temperature.

2.2 Loss of Secondary Coolant

The simulation of loss of secondary cooling with a heating rate for bulk sodium of 0.6°F/s (0.33°C/s) indicates that loss of secondary cooling would have little effect upon reactor-coolant or fuel-element temperatures. The power history (see Fig. 63) indicates an 18% reduction in reactor power during the 80 s of simulation. The reduction is due to the negative thermal-reactivity coefficient of EBR-II. The plots of fuel and coolant temperatures during the transient are shown in Figs. 64-66. These plots indicate a general trend toward a slight temperature increase during the first 30 s of the transient, followed by a decreasing temperature for the remainder of the transient.

The plot of the temperature of the hottest fuel surface (Fig. 65) indicates an initial temperature of 1075°F (579°C), an insignificant rise during the first 10 s, and a minimum temperature of 1060°F (571°C) at 80 s. Uncertainty factors are of little interest here, because the normal temperatures are, for practical purposes, the maxima.

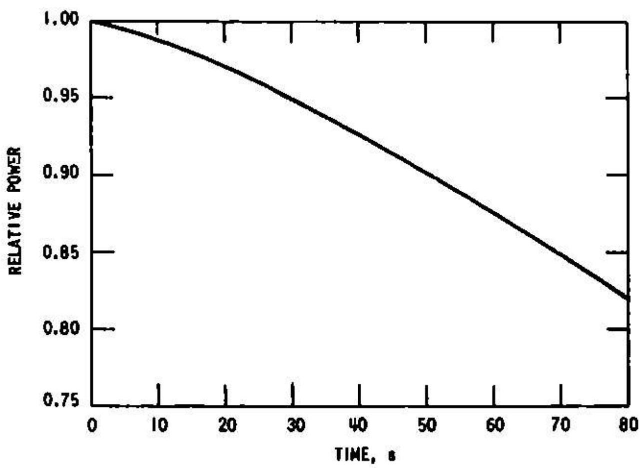


Fig. 63. Power History for Analysis of Loss of Secondary Flow

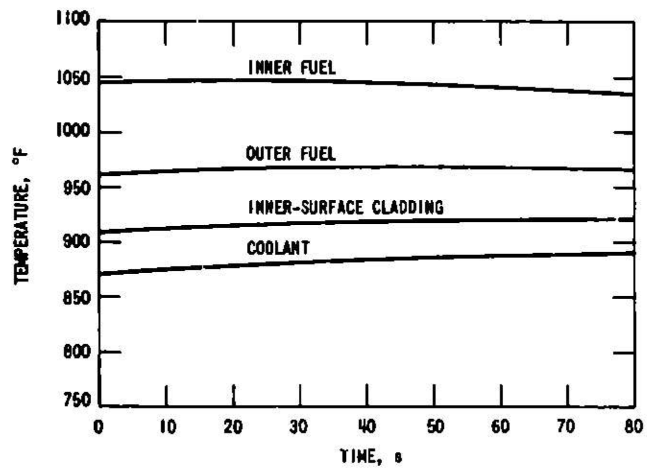


Fig. 64. Temperature of Average Driver-fuel Element in Row 2: Loss-of-secondary-flow Analysis. Conversion factor: °C = (°F - 32)/1.8.

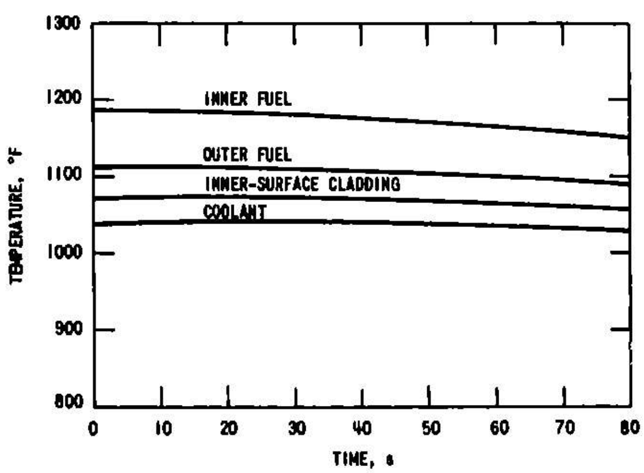


Fig. 65. Temperature of Hottest Driver-fuel Element in Row 6: Loss-of-secondary-flow Analysis. Conversion factor: °C = (°F - 32)/1.8.

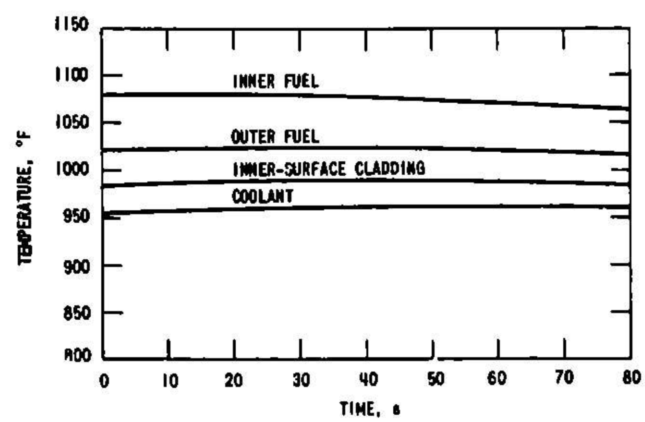


Fig. 66. Temperature of Average Driver-fuel Element in Row 6: Loss-of-secondary-flow Analysis. Conversion factor: °C = (°F - 32)/1.8.

If the operator does not intervene during a loss of seconday cooling by maintaining core ΔT, the core ΔT would decrease because of negative feedback That is, the subassembly outlet coolant temperature would not rise as fast as the inlet coolant temperature, and the operator could respond by increasing reactor power. This action would continue until the reactor shuts down because of high subassembly outlet temperature. At a heating rate of 0.6°F/s (0.33°C/s) for reactor inlet temperature, if the operator maintains specified core ΔT, the subassembly outlet coolant temperature would rise in direct response to the rate of heating of inlet coolant. The worst-case trip value for the subassembly outlet temperature would occur in about 50 s. At this time, the bulk-sodium temperature would be about 730°F (388°C) and the maximum fuel temperature would be about 1105°F (596°C). With uncertainty factors, the maximum fuel temperature would be no greater than 1169°F (632°C). These values are well within the safety limits as listed in Sec. 2.1 of this appendix.

3. Sensitivity Analysis

The results of the analyses of overcooling and undercooling of the primary sodium presented in previous sections were based on the characteristics of a specific reactor loading considered to represent present and planned reactor configurations. The sensitivity of the results of these analyses to changes in reactivity coefficients has been considered; Fig. 67 shows the relationship between outlet temperature and inlet temperature for bounding values of inlet-temperature coefficients. The behavior of the system with a hypothetical zero inlet-temperature coefficient would be independent of the isothermal coefficient. The data for the limiting case in which the inlet-temperature coefficient equals the isothermal coefficient are based on the normal value of 1 Ih/°F (1.8 Ih/°C) for the isothermal coefficient determined from measurements on several EBR-II core loadings.

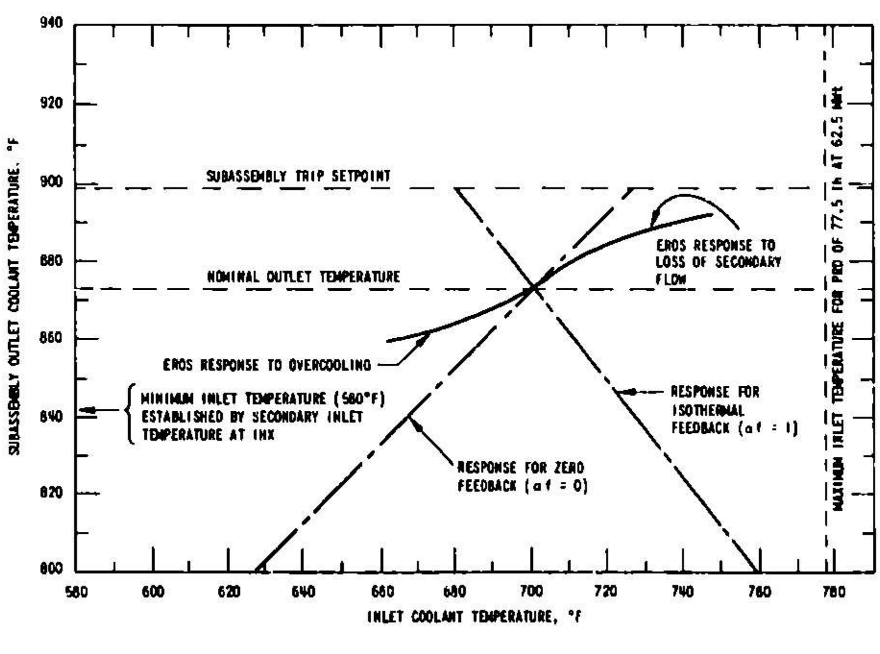


Fig. 67. Temperature of Subassembly Outlet Coolant as a Function of Inlet Coolant Temperature. Conversion factor: °C = (°F - 32)/1.8.

For zero feedback, reactor power would remain constant, in the absence of any operator action, and the outlet temperature would increase or decrease in step with the inlet temperature. The maximum temperature would be limited to a safe value by the trips for subassembly outlet temperature. If the operator increased power, the power would be limited to a safe value by the overpower trips.

Because EBR-II has a negative inlet-temperature coefficient, the reactor power would decrease with increasing inlet temperature, in the absence of any operator action, until the reactor was critical at the temperature at which isothermal feedback and power-reactivity decrement (PRD) were equal. This point would be independent of the actual value of negative inlet-temperature coefficient. If the operator added reactivity to maintain power, the reactor would be protected by the trips for subassembly outlet temperature. Conversely, a decreasing inlet temperature would cause an increase in power, and the reactor would be protected by the trips caused by overpower.

The PRD and the isothermal temperature coefficient will tend to change in the same direction (see Table XIII), but not proportionately. Available experimental data show that the inlet temperature at which the reactor would be at zero power and critical during a loss of secondary-system cooling, without operator action, could be 760-810°F (404-432°C). This temperature is within the range considered safe for both the reactor fuel and the primary tank.

The "EROS response" data shown in Fig. 67 are from an 80-s transient calculation made under the limiting-case assumption that the rate of change of inlet temperature would remain constant at the initial full-power value. The actual temperature rate would be a function of reactor power and would produce less rapid changes in temperature and power than those calculated.

The EROS calculation does not model reactivity effects of temperature changes in the reactor inlet plenum, lower reflector, or grid plate. The net result of neglecting inlet effects is that the EROS results approximate what would be expected if the isothermal coefficient were about 0.43 Ih/MW. Therefore, the "EROS response" curve in Fig. 63 cannot be extrapolated to obtain the reactor-shutdown temperature for loss of secondary flow.

In summary, the behavior of the reactor during conditions under which the primary system is undercooled or overcooled by the secondary system depends on the inlet-temperature coefficient, the isothermal temperature coefficient, and the PRD. For ranges of these variables considered credible for EBR-II with metal driver fuel and a stainless steel reflector, the reactor will be protected from damage by the combination of power-level trips, subassembly-outlet-temperature trips, and the required negative temperature coefficient.

APPENDIX G

Thermal Effects of Sodium Burning in Reactor Containment Building

The worst-case sodium spill in the reactor containment building is a postulated break in the primary purification cell.[30] The sodium and NaK inventory of that system is about 550 gal (2.08 m^3), of which 328 gal (1.24 m^3) is sodium.

The other systems that carry sodium outside of the primary tank are the radioactive sodium chemistry loop (RSCL) and the fuel-element rupture detector (FERD). Neither system contains a sodium inventory to match that of the primary purification cell. A leak of the 2-in. (5-cm) stainless steel line downstream of the FERD EM pump would result in air inleakage because the system operates at subatmospheric pressure. (Note that the volume of sodium spilled affects the duration of burning but not the heat-up rate, because the burning rate is controlled by the burning-surface area, sodium purity, and the available air supply.)

Sodium burning has also been hypothesized within the primary tank associated with air inleakage to the argon cover gas. (See Appendix C.) The worst case was determined to be associated with a breakage of the inlet line of the argon cooling system (ACS) for the fuel-unloading machine (FUM) when in communication with the argon cover gas. It was then assumed that air was pumped at the maximum rate of the FUM-ACS into the primary tank. The resulting sodium-pool burning would have no effect on reactor-building temperature.

Reference 1 evaluated sodium burning in the reactor containment building because of spray exposure of primary sodium. No mechanism was identified for such a fault, which could be hypothesized to occur only after a major reactor accident. A reactor trip resulting from containment-building isolation would have no effect in such a circumstance and is not a consideration here.

To analyze the worst-case sodium leak, 328 gal (1.24 m^3) of sodium are assumed to be spilled over an available area of 74 ft^2 (6.88 m^2). The average resulting burning rate for such circumstances is 10 $lb/ft^2{\cdot}h$ (0.014 $kg/m^2{\cdot}s$), with a burning rate of 740 lb/h (0.093 kg/s). The total heat released from complete combustion of sodium is 4500 Btu/lb (1.046 x 10^7 J/kg), with a resulting energy release rate of 55 000 Btu/min (58 MJ/min).

With data from the test of sodium-pool burning, the fraction of energy released to the surrounding air may be calculated as follows:

From HEDL test F-1:

Initial burning rate	= 7.7 $lb/h{\cdot}ft^2$ (0.626 $kg/m^2{\cdot}s$)
Burn-pan surface area	= 31.5 ft^2 (2.93 m^2)

Volume of enclosure = 3350 ft^3 (94.8 m^3)

Heating rate = 7.7 x 31.5 x 4500 = 1 090 000 Btu/h (1.15 GJ/h)

= 18 200 Btu/min (19.2 MJ/min)

Measured heating rate of air (for first 10 min) = 13°F/min (7.2°C/min)

Mass of air in enclosure = 3350 ft^3 (94.8 m^3)

Heat-input rate = mc_pT = 252 x 0.241 x 13 = 790 Btu/min (0.8 MJ/min)

Fraction of heat transferred to air = 790/18 200 = 0.043.

The rate of temperature rise in the reactor containment building may be calculated as follows:

Volume of air in the reactor containment building = 456 000 ft^3 (12 904 m^3)

Mass of air = 456 000 x 0.07528 x 12.2/14.7

= 28 490 lb (12 922 kg) [at 12.2 psia (84.1 kPa) and 68°F (20°C)].

The mean rate of temperature rise (assuming the heat is uniformly distributed) would be

$$\frac{\Delta\text{temp}}{\Delta\text{time}} = \frac{(55\ 000\ \text{Btu/min})\left(0.043\,\dfrac{\text{heat to air}}{\text{total heat}}\right)}{(28\ 490)\ \text{lb}_m \times 0.241\,\dfrac{\text{Btu}}{\text{lb}_m\,{}^\circ\text{F}}}$$

$$= 0.34^\circ\text{F/min}\ (0.19^\circ\text{C/min}).$$

For this calculation, uniform heating of the containment-building atmosphere is assumed; such uniformity would not occur with actual burning in the primary purification cell. The actual rate of temperature rise as recorded by thermocouples (which are part of the isolation system) would be less than the average.

REFERENCES

1. L. J. Koch et al., *Hazard Summary Report, Experimental Breeder Reactor II (EBR-II)*, ANL-5719 (May 1957).
2. L. J. Koch et al., *Addendum to Hazard Summary Report, Experimental Breeder Reactor-II (EBR-II)*, ANL-5719 (Addendum) (June 1962).
3. *Supplement CH-CA0540 to ERDA Manual Chapter 0540*, "Safety of ERDA-owned Reactors (Feb 4, 1975).
4. *Supplementary Criteria for RDT Reactor Plant Protection Systems*, RDT C16-1T, U. S. Atomic Energy Commission (Dec 1969).
5. R. N. Curran et al., *Design Basis Document for Trips Related to Reactivity Change in the EBR-II Plant Protection System (in the Operate Mode)*, ANL-76-32 (to be published).
6. E. M. Dean and J. I. Sackett, *Response of EBR-II to Off-normal Reactivity Insertion*, ANL-75-41 (to be published).
7. System/360 Continuous Modeling Program (360A-CX-16X), IBM Application Program H20-0367-2 (1968).
8. *Nuclear Instrumentation, Simulation and Control*, Gulf General Atomic Inc. (Sept 1969).
9. *Wide-range (10 decade) Neutron Flux Monitoring Channel*, RDT C15-2T (Feb 9, 1971).
10. J. F. Boland et al., *Design Basis Document for Trips Related to Loss of Primary Flow in the EBR-II Plant Protection System*, ANL-76-31 (to be published).
11. J. F. Boland, *Nuclear Reactor Instrumentation (In-Core)*, Gordon and Breach, New York (1970).
12. S. T. Zegler et al., *Compatibility of Uranium-5 w/o Fissium Alloy with Types 304L and 316 Stainless Steel*, ANL-7596 (Sept 1969).
13. C. M. Walter and L. R. Kelman, *The Interaction of Iron with Molten Uranium*, J. Nucl. Mater. *20*, 314-322 (1966).
14. A. B. Rothman et al., *Transient Behavior of Pre-Irradiation "High Swelling" EBR-II Driver Fuel in TREAT*, Trans. Am. Nucl. Soc. *12*(2), 867 (Nov 1969).
15. R. A. Cushman, B. R. Seghal, and V. W. Lowery, *Prospectus for Operating EBR-II with High-worth Control Rods*, ANL/EBR-066 (Sept 1972).
16. W. E. Vesely and R. E. Narum, *PREP and KITT: Computer Codes for the Automatic Evaluation of a Fault Tree*, IN-1349 (Aug 1970).
17. *ASME Boiler and Pressure Vessel Code*, Section VIII (1974).
18. *ASME Boiler and Pressure Vessel Code*, Section III (1974).
19. R. K. Hillyard and J. M. Yatabe, *FFTF Secondary Sodium Fire Protection System Test F1*, HEDL-TIME-73-48 (Apr 1970).
20. J. J. Kaganove, E. M. Dean, and A. V. Campise, *EBR-II Reactor Operation Simulator Volume I, A Description of EROS*, ANL-7981 (Aug 1974).

21. A. V. Campise, *Dynamic Simulation and Analysis of EBR-II Rod-drop Experiments*, ANL-7664 (Mar 1970).

22. *Bettis Technical Review*, WAPD-BT-24 (Dec 1961).

23. D. W. Cissel, L. F. Coleman, F. O. Pancner, F. A. Smith, and A. D. Tevebaugh, *Guidelines for Sodium Fire Prevention, Detection, and Control*, ANL-7691 (June 1970).

24. J. D. Gracie and J. J. Droher, *A Study of Sodium Fires*, NAA-SR-4383, Atomics International.

25. S. Timoshenko and S. Woimonsky-Krieger, *Theory of Plates and Shells*, 2nd ed., McGraw-Hill, New York (1959).

26. L. K. Chang and M. F. Card, *Thermal Buckling Analysis for Stiffened Orthotropic Cylindrical Shells*, NASA-TND-6332 (Apr 1971).

27. D. L. Block, *Influence of Discrete Ring Stiffeners and Prebuckling Deformation on the Buckling of Eccentrically Stiffened Orthotropic Cylinder*, NASA-TND-4283 (1968).

28. S. B. Batdorf and M. Schildcrout, *Critical Axial-compressive Stress of a Curved Rectangular Panel with a Central Chordwise Stiffener*, NACA-TN-1661 (1948).

29. R. W. Seidensticker, *Design, Analysis, and Construction of EBR-II Primary Tank and Associated Structures*, ANL-6922 (Sept 1970).

30. C. C. Price, J. A. Bjorkland, and W. F. Booty, *Basis for the EBR-II Reactor Containment Criteria*, ANL-76-33 (to be published).