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GA-10298
Amendment 10

GAS-COOLED FAST BREEDER REACTOR PRELIMINARY SAFETY INFORMATION DOCUMENT AMENDMENT 10

GCFR RESIDUAL HEAT REMOVAL SYSTEM CRITERIA, DESIGN, AND PERFORMANCE

Prepared under
Contract DE-AT03-76SF71023
for the San Francisco Operations Office
Department of Energy

DATE PUBLISHED: SEPTEMBER 1, 1980

GENERAL ATOMIC COMPANY

DISCLAIMER

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FOREWORD

The objective of this report, PSID* Amendment 10, is to present the comprehensive set of safety design bases for the conceptual design of the gas-cooled fast breeder reactor (GCFR) residual heat removal (RHR) systems in a manner which will enable the Nuclear Regulatory Commission (NRC) to review and comment on licensability of these design bases. This report also presents information concerning a specific plant design and its performance as an auxiliary part to assist the NRC in evaluating the safety design bases. The NRC is not requested to review and concur with the design and performance data, although a dialog with the NRC in these areas is desirable.

Since the last PSID Amendment was issued, major design revisions, such as a natural circulation RHR capability, have been adopted. This amendment was prepared in the form of a self-contained document which contains a complete set of safety design bases, a description of the updated GCFR demonstration plant, and an evaluation of the core cooling performance using the revised RHR systems.

*"Gas-Cooled Fast Breeder Reactor Preliminary Safety Information Document," General Atomic Report GA-10298, February 15, 1971.

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FIGURES (continued)

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TABLES (continued)

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1. SUMMARY AND CONCLUSIONS

This report presents a comprehensive set of safety design bases to support the conceptual design of the gas-cooled fast breeder reactor (GCFR) residual heat removal (RHR) systems. The report is structured to enable the Nuclear Regulatory Commission (NRC) to review and comment on the licensability of these design bases. This report also presents information concerning a specific plant design and its performance as an auxiliary part to assist the NRC in evaluating the safety design bases. The NRC is not requested to review and concur with the design and performance data, although a dialog with the NRC in these areas is desirable.

1.1. SUMMARY OF GCFR BACKGROUND

The GCFR development was initiated at General Atomic (GA) in the early 1960s. GCFR development work has been contributed by U.S. national laboratories and by European industries and governmental agencies. GCFR development has been supported by both the U.S. government through the Department of Energy (DOE) by GA and by utilities through Helium Breeder Associates (HBA).

Evolving from the early work, the GCFR Preliminary Safety Information Document (PSID) was submitted to NRC in 1971 to serve as a basis for an information exchange in evaluating the GCFR concept (Ref. 1-1). Summarizing the initial phase of its review, the NRC issued a Preapplication Safety Evaluation of the GCFR in 1974 (Ref. 1-2). Section 2.1 details the GCFR development background.

Recently, the plant design has been greatly revised for improved core cooling reliability and performance. As a result, several different RHR systems, including forced circulation and natural circulation configurations, are now available to the GCFR.

1.2. SUMMARY OF SAFETY DESIGN BASES FOR RHR

Safety design bases for the GCFR RHR are derived to adequately assure that acceptable fuel cladding and pressure boundary temperatures are maintained for all credible events which lead to reactor shutdown. Parallel references are made to NRC licensing criteria and positions for the light water reactor (LWR) and the liquid metal fast breeder reactor (LMFBR), particularly the Clinch River Breeder Reactor (CRBR) (Ref. 1-3), in developing the GCFR safety design bases for RHR.

Key elements of the GCFR safety design bases follow:

1. Two redundant safety systems are to be provided for long-term RHR: (a) the core auxiliary cooling system (CACS) and (b) the shutdown cooling system (SCS).

2. Both the CACS and the SCS are seismic category I.

3. The SCS and the CACS shall be independent from each other.

4. The reliability goal for the RHR function shall be such that the probability of loss of design core cooling geometry shall be beyond the design basis value.

5. Natural circulation RHR capability shall be provided with appropriate experimental verification of natural circulation performance.

The requirement of two safety RHR systems enhances safety and reliability in excess of the Atomic Energy Commission (AEC) General Design Criteria (GDC) 34 and 35 (Ref. 1-4) for LWRs, which require one safety system for RHR. It is also consistent with the NRC position on fast reactor licensing applied to the CRBR in 1976. NRC concurrence with these GCFR RHR safety design bases is requested as part of the review of this report.

1.3. SUMMARY OF GCFR DEMONSTRATION PLANT DESCRIPTION

The GCFR demonstration plant employs pressurized helium as the reactor primary coolant. A prestressed concrete reactor vessel (PCRV) contains the reactor core, the steam generator, and the helium circulator in each of the three main cooling loops and the auxiliary circulator and the heat exchangers in each of the three CACS loops. The main cooling loops provide reactor cooling during power operation, and one of several RHR systems performs decay heat removal. The reactor core consists of hexagonal fuel and blanket assemblies. Each assembly contains a large number of fuel rods which are similar to LMFBR fuel rods, except that the stainless steel cladding surface is roughened to improve heat transfer and the rods are vented to the primary coolant by means of a pressure equalization system (PES). The PES continuously removes fission gas from each of the fuel rods through the vent channels. Heat from the fuel rods is transferred to the helium coolant, which transports the heat to the steam generator. Steam from the steam generator generates electricity through a balance-of-plant (BOP) arrangement similar to those used in other nuclear- or fossil-powered plants. Section 4 details the GCFR plant.

Major design revisions for important aspects of the current demonstration plant design which have been implemented since the previous PSID design are as follows:

1.3.1. Upflow Core with Natural Circulation RHR

The primary coolant flow direction has been reversed from downflow to upflow through the core. This provides a natural circulation capability, a substantial safety asset. The natural circulation provides diversity to the forced circulation systems and an inherently passive and long-term RHR capability with minimum operator or powered actions.

1.3.2. Incorporation of Multiple Safety RHR Systems

The plant design incorporates a safety-class RHR system, the SCS, in addition to the existing safety-class CACS and the nonsafety-class main loop cooling system (MLCS).

1.3.3. Electrically Driven Radial-Flow Main Circulators

The electrically driven radial-flow circulators provide the following benefits over the steam-driven axial-flow circulators used in the previous demonstration plant.

1. Simpler control due to system decoupling between the heat source (reactor) and the drive power (circulator).

2. Higher stall resistance of radial-flow over axial-flow circulators.

3. Longer inertial coastdown due to a massive electric motor versus the compact steam turbine.

4. Preoperation testing ease in providing the circulator power using off-site ac power instead of high-pressure, high-flow steam required in case of the turbine drive.

1.3.4. Low Core Flow Pressure Drop

The core pressure drop was decreased from the previous demonstration plant value described in Ref. 1-5 by approximately a factor of two. This design revision relaxes the coolant circulator requirement significantly under accident conditions and enhances the GCFR natural circulation capability.

1.3.5. RHR Systems

The GCFR has the following four methods of RHR available:

1.3.5.1. MLCS, Nonsafety Class. The MLCS RHR mode (see Section 4.5.1) is obtained by the main loops continuing to operate after reactor shutdown. Following a reactor trip, the main circulator speed is reduced to a shutdown cooling level (typically 30% for pressurized coolant and higher speeds that are inversely proportional to the coolant pressure at depressurized conditions) and the main turbine-generator is tripped, diverting the steam to the desuperheater. Concurrently, the feedwater flow is ramped to the shutdown rate (typically 25%). Eventually, the steam generators are fully flooded, and the feedwater flow, steam generator pressure, and helium flow are adjusted for long-term RHR. This system is available for all shutdown core cooling needs with off-site power at either pressurized or depressurized coolant conditions, as long as off-site power is available.

1.3.5.2. SCS, Safety Class. The SCS (see Section 4.5.2) consists entirely of safety-class equipment. The SCS shares the main circulator, the circulator shaft, and the steam generator with the MLCS. The SCS uses a pony motor to drive the circulator with safety class (1E) power.

The SCS will be used for RHR under a number of accident conditions, such as loss of off-site power (LOSP), loss of feed-water, etc. Operation is transferred from the MLCS to the SCS as follows:

1. When the plant protection system (PPS) initiates the SCS, the pony motors are energized to maintain the circulator speed, and the SCS heat rejection system is activated.

2. The feedwater flow from the boiler feed pump (BFP) is isolated and replaced by recirculation from the SCS heat rejection condenser.

3. Using the circulating water pump, the water is circulated in a closed loop through condenser tubes submerged in an atmospheric-pressure water tank, where the core decay heat is removed by heating and evaporating the water.

1.3.5.3. CACS, Safety Class. The CACS (see Section 4.5.3) is the most comprehensive RHR system available for pressurized and depressurized coolant conditions. Each of three CACS loops is comprised of an electrically driven auxiliary circulator, a check valve, and a helium-to-water heat exchanger, the core auxiliary heat exchanger (CAHE). The water from the CAHE is circulated through a pressurized water loop with pumps and a pressurizer, and the heat is rejected to the atmosphere by air from fans in the auxiliary loop cooler (ALC), a finned tube heat exchanger.

1.3.5.4. Natural Circulation CACS, Safety Class. The CACS design incorporates natural circulation capabilities on the helium, water, and air sides as a backup to normal forced circulation capabilities. Using the CACS, core decay heat is transported by the primary coolant helium to high-pressure water in the CAHE, which is elevated above the core. Heated water from the CAHE reaches the ALC, located above the CAHE, by natural circulation in the pressurized water loop. The heat from the ALC is ultimately rejected to the atmoshpere by natural air draft through a tall chimney. Natural circulation core cooling is available for an indefinite period after a total loss of forced circulation (LOFC) capability with the primary coolant pressurized. With depressurized coolant, natural circulation in the primary loops is not adequate, but natural circulation in the secondary water and the tertiary air are available as backup to forced circulation. A repressurization feature for the primary coolant is incorporated to induce adequate natural circulation for mitigating a complete LOFC under refueling conditions.

1.3.6. RHR System Operation

The logic for selecting and initiating the RHR systems (see Section 4.5) is based on using operating equipment before switching to equipment that must be brought into operation. The normal sequence of RHR operation is (1) MLCS, (2) SCS, (3) forced circulation CACS, and (4) natural circulation CACS. However, if an event cannot be accommodated properly by the SCS system, the RHR initiation system will start up CACS loops and, upon verifying CACS operation, will shut down all SCS loops.

The SCS and CACS are intended to be independent and diverse safety systems. They are mechanically and electrically isolated; they are powered by different and diverse IE power systems to make the systems independent and to increase their resistance to common mode failures. Diversity is employed to make the system less vulnerable to common cause failures. The CACS and SCS are designed to meet all the requirements of Institute of Electrical and Electronic Engineers [i.e., IEEE 279 (Ref. 1-6) and IEEE 603 (Ref. 1-7)]; Regulatory Guide 10CFR50, Appendix A (Ref. 1-4); and NRC Regulatory Guides related to safety systems.

1.4. SUMMARY OF CORE COOLING PERFORMANCE

1.4.1. Selection of Transients

This report selects and analyzes categories of transient events particularly important in determining GCFR RHR system adequacy:

1. Decrease in reactor primary coolant flow rate.
2. Decrease in reactor heat removal by the secondary system.
3. Decrease in reactor coolant inventory.
4. Reactivity accidents.

The initiating events in each of the above categories are further classified into five American Nuclear Society (ANS) plant conditions (PCs). These are based on the principle that the most probable occurrences should be accommodated by the largest design margin and yield the least consequence and that those extreme situations having the potential for the greatest risk should be those least likely to occur.

In selecting a sequence of events to be used for the analysis, the deterministic safety evaluation rules of ANS-50, Policy 2.4 (Ref. 1-8) were used. These rules require postulated failures to be consistently applied to meet a consistent set of safety criteria when the plant responds to the initiating event. These rules also govern applying the single failure to safety-class components and applying the coincident occurrence to nonsafety components. In general, when either the single failure or the coincident occurrence is assumed in addition to the initiating event, the safety limits of the next higher PC are allowed. The rules also require that the event scenarios combining a single failure and a coincident occurrence be considered, but with PC-5 limits not to be exceeded.

Wherever possible, the transient analyses are presented in a manner conforming to the standard format and content for a safety analysis report (SAR) set out in Regulatory Guide 1.70 (Ref. 1-9).

Results of the deterministic evaluation of the plant response to transient events indicate that only a part of the four RHR systems is needed; other available systems are retained without being used unless multiple failures are postulated (see Section 5).

To indicate a large margin in the GCFR RHR capability, cases of plant response to multiple failures beyond the deterministic rules are analyzed in addition to the deterministic transient scenarios.

1.4.2. Design Limits

Section 5.1.2. presents the design temperature limits for the core and essential primary loop components for various PCs which depend on expected frequency of occurrence. Under PC-5 (i.e., faulted condition), adequacy of the RHR capability is determined when the transient temperatures meet the temperature limits of 1300°C (2372°F) for the fuel and blanket rod claddings, 2800°C (5070°F) for the fuel rod centerline melting, and 980°C (1800°F) for the PCRV thermal barriers. For other PCs, lower temperature limits apply.

1.4.3. Plant Characteristics

Section 5.1.3. presents the plant characteristics data used in the cooling performance evaluation, including the conservative plant initial conditions, the core and the blanket power distribution, and the control and shutdown rod insertion characteristics for primary and secondary reactor trip systems. It presents uncertainty margins for the system performance parameters used for conservative accident analyses. The most significant uncertainties are the 20% decay heat and 20% coolant flow pressure drop.

The core decay heat correlation is based on ANS Standard 5.1 (Ref. 1-10). Since the blanket RHR is imporant in many cases, Section 5.1.4. develops and presents the blanket decay heat correlation with a major contribution of the core gamma ray transport. It also presents the gamma heating in fuel assembly duct wall, important in the RHR phase.

Section 5.1.5. describes the assumed protection actions by the reactor trip systems and the RHR initiation system. It presents the setpoints for reactor trip parameters with their time delays. The RHR initiation system commands use of one of the three forced-circulation RHR systems (i.e., MLCS, SCS, and CACS) according to their availability and the plant RHR need.

1.4.4. Methods of Analysis

Section 5.1.6. briefly describes computer programs used. FASTRAN is a generic GCFR system dynamics program used for accident analyses. RATSAM is a generic system dynamics program for gas-cooled reactor plants. RATSAM lacks modeling of some components, but it can deal with unequal loops; therefore, it is used primarily for natural circulation analyses with various loop conditions. The CNTB program calculates the pressure and temperature responses of the reactor containment building atmosphere during a postulated depressurization accident. The COBRA program is used for subchannel thermal-hydraulic analysis for the fuel and blanket assemblies.

1.4.5. Results of RHR Performance Analysis

Section 5 summarizes results of the transient analyses for the four key accident event categories. Sections 5.2 through 5.5 examine the plant responses and the performance capability of the RHR systems under accidents of these categories. The analyses are performed in two stages in each of the categories.

In the first stage, the event sequences are selected and analyzed according to the deterministic criteria. Results of the deterministic events indicate that the RHR capability is adequate to meet the limiting temperatures of the core and the essential components with significant margins in meeting the temperature limits.

In the second stage, the margin cases are defined by assuming multiple failures beyond the deterministic event sequence and are analyzed to examine the capability of all available redundant RHR systems.

Since the GCFR design base requires that adequate core cooling be provided by one of the several forced circulation RHR systems under all the design basis events, application of the deterministic criteria only will never lead to the events using natural circulation RHR which are described in the category of LOFC (Section 5.6). Therefore, all the natural circulation events following LOFC are considered margin cases.

Section 5.7 summarizes the core cooling performance evaluation. Examination of the transient results for all cases indicates that the component limits are met as long as the core cladding temperature meets its limits, except for a few over-power cases, where the fuel centerline temperature is more limiting.

Section 5.7.1 discusses the core cooling performance margin. Assessment of an accurate margin depends on the uncertainty treatment in the analyses. Section 5.7.1 chooses the DBDA as the most limiting case of the core cooling and explores the core cooling margins using the cumulative and statistical uncertainty combinations and the best estimate model without uncertainties. The conservative model with cumulative uncertainties indicates a 200°C (360°F) margin to the core cladding damage limit. This margin adequately allows for the local excess temperatures occurring at the fuel assembly edges.

A more realistic margin of 719°C (1294°F) is obtained by statistically combining system parameter uncertainties, while a 837°C (1507°F) margin is obtained by using the best estimate model without uncertainties but with a single loop failure.

Section 5.7.2 discusses depth of protection available beyond the deterministic requirements. Section 5.7.2 summarizes the key results of all the RHR cases analyzed to indicate how the fuel and blanket cladding temperatures meet the respective limits at various PCs and shows maximum fuel and blanket cladding temperatures against the number and type of RHR systems. It demonstrates not only adequate core cooling with respect to the design temperature limits, but also a significant depth of protection with multiple RHR system redundancy.

1.5. SUMMARY OF RHR VERIFICATION PLAN

Some uncertainties exist in predicting RHR performance, because transient thermal-hydraulic effects in the coolant systems are complex. Therefore, the predictive methods and the RHR performance of the GCFR coolant systems will be systematically verified.

Section 6 outlines key elements of two separate plans to verify each operation mode. Section 6.1 addresses verifying GCFR-RHR capability; it focuses on the deterministic requirements that are met by forced circulation RHR systems. Section 6.2 describes a similar plan developed to verify and validate the GCFR natural circulation RHR capability, which is considered to provide an added margin for events beyond the design basis.

Both plans contain the following general task elements which will develop confidence in the system and component designs and in their performance predictions:

1. Survey RHR systems and performance verification for other reactors.

2. Identify key issues for GCFR RHR system performance.

3. Verify by comparing to independent codes.

4. Validate using data from other reactors.

5. Validate by component and subsystem tests.

6. Validate by preoperational and startup tests.

7. Investigate RHR system adequacy for postulated event sequences.

1.6. CONCLUSIONS

1. Safety design bases for the GCFR RHR systems provide a margin of safety in excess of the minimum requirements for the General Design Criteria for the LWR and comply with the NRC position relative to LMFBRs.

2. Performance evaluation for the GCFR RHR systems indicates that these systems not only fully meet the safety design bases, but also provide significant margins in their capacities and availability as redundant systems.

3. The natural circulation RHR redundant system is particularly significant, because it provides inherently passive and diverse backup to the forced circulation system whenever the reactor is pressurized.

REFERENCES

1-1. "Gas-Cooled Fast Breeder Reactor Preliminary Safety Information Document," General Atomic Report GA-10298, February 15, 1971.

1-2. "Preapplication Safety Evaluation of the Gas-Cooled Fast Breeder Reactor," U.S. Atomic Energy Commission Directorate of Licensing, August 1, 1974.

1-3. Denice, R., letter to Clinch River Breeder Reactor Project, May 6, 1976.

1-4. "General Design Criteria for Nuclear Power Plants," Code of Federal Regulations, Title 10, Part 50, Appendix A, U.S. Government Printing Office, Washington, D.C. 1977.

1-5. "Gas-Cooled Fast Breeder Reactor Preliminary Safety Information Document," General Atomic Report GA-10298, February 15, 1975 (Amended 1976).

1-6. The Institute of Electrical and Electronic Engineers, Inc. (IEEE) Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

1-7. The Institute of Electrical and Electronic Engineers, Inc. (IEEE) Standard 603-1980, "Criteria for Safety Systems for Nuclear Power Generating Stations."

1-8. "Plant Design Conditions for Nuclear Power Generating Stations," American Nuclear Society ANS-50, Policy 2.4, June 1978.

1-9. "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Nuclear Regulatory Commission Regulatory Guide 1.70, Revision 3, November 1978.

1-10. "Decay Energy Release Rates Following Shutdown of Uranium Fueled Thermal Reactors," American Nuclear Society Proposed Standard 5.1, October 1971.

2. INTRODUCTION

This report presents a comprehensive set of safety design bases for the conceptual design of the gas-cooled fast breeder reactor (GCFR) residual heat removal (RHR) systems in a manner and format to enable the Nuclear Regulatory Commission (NRC) to review and comment on the licensability of these design bases. This report also presents information concerning a specific plant design and its performance as an auxiliary part to assist the NRC in evaluating the safety design bases. The NRC is not requested to review and concur with the design and performance data, although a dialog with the NRC in these areas is desirable.

The GCFR has been under development since the early 1960s, and technical information has been exchanged with the U.S. Atomic Energy Commission (AEC) Directorate of Licensing (DOL) and the Advisory Committee on Reactor Safeguards (ACRS). Section 2.1 briefly describes the GCFR development background. Later sections also specify the report scope and the requested NRC actions following review of this report.

The central part of this report describes the safety design bases for the core cooling system presented in Section 3. These design bases are the backbone of GCFR safety and, therefore, are to be examined with a view to determining their suitability and adequacy for eventual plant licensing.

Because the RHR systems are only part of the total GCFR plant, information is presented on the conceptual design of the overall plant and interfacing systems. Recent major design revisions have impacted the RHR operations. A conceptual design of the GCFR demonstration plant includes these design revisions (see Section 4).

To determine whether the conceptual GCFR design can meet the safety design bases of Section 3, Section 5 evaluates the core cooling performance with respect to anticipated and accident events.

The natural circulation core cooling capability is an important RHR feature. It is a redundant, diverse, and passive mode of operation requiring no operator or powered action. Due to the importance of the RHR function, an extensive verification program for RHR performance and its prediction methods is under way (see Section 6).

2.1. BACKGROUND OF GCFR DEVELOPMENT

The GCFR development program was initiated at General Atomic Company (GA) in the early 1960s. The program has been supported by both government and private funding. In 1968, a large number of U.S. utility companies formed the GCFR Utility Program to increase financial support and to ensure that the utility user interests were considered in GCFR design and development. In 1976, the electric utilities organized Helium Breeder Associates (HBA), a nonprofit corporation, to manage GCFR development for the end-user.

The GCFR program has also obtained considerable internal cooperation from the German national laboratories at Karlsruhe (KfK) and Jülich (KfA); the German nuclear supplier, Kraftwerk Union (KWU); and the Swiss National Laboratory for Reactors at Wuerenlingen (EIR). An independent study program of a commercial-size GCFR plant was begun in the late 1960s under the auspices of the European Association for Gas-Cooled Breeder Reactors (GBRA) in Brussels, Belgium.

Since the late 1960s, the major effort of the U.S. GCFR program has been directed toward developing a GCFR demonstration plant design. By early 1971, a conceptual design was developed and information relevant to the safety characteristics of a 300-MW(e) demonstration plant was documented in a Preliminary Safety Information Document (PSID) (Ref. 2-1).

The PSID provided a basis for a preapplication safety evaluation of the GCFR demonstration plant by the AEC DOL and the ACRS. During the period from 1971 through 1974, a series of technical exchanges took place with the AEC DOL and the ACRS. These exchanges were to acquaint the AEC with the safety-related design bases and design features of the GCFR and to assure their suitability and adequacy for eventual plant licensing. The exchanges took the form of formal written questions following meetings or reviews of the GA submittals. Responses to these questions were issued in a series of Supplements and Amendments to the PSID (Refs. 2-2 through 2-4), together with design changes made during the years of the licensing review. Responses to a total of 209 AEC DOL and 33 ACRS questions were submitted.

A preliminary environmental report was also prepared and submitted to the AEC DOL for review (Ref. 2-5), and the ACRS established a GCFR subcommittee to review the GCFR design described in the PSID. In August 1974, the AEC DOL issued a Preapplication Safety Evaluation Report (PSER) (Ref. 2-6), and the ACRS issued an interim letter (Ref. 2-7). The PSER identified several areas requiring additional work but concluded that the proposed demonstration plant, as conditioned by their report, could potentially be operated without undue risk to the health and safety of the public. The interim ACRS better recognized certain advantageous safety characteristics of the GCFR (principally those associated with the reactivity, chemical, and maintenance advantages of the helium coolant) and identified several areas which required more work. Subsequent work on the GCFR program has been directed toward additional development work and design modifications in response to the concerns listed by the AEC DOL and the ACRS.

Recently, GA conducted a major review of alternate plant designs to improve the reliability and adequacy of reactor core cooling. Major design improvements studied and adopted are the following:

1. An upflow core with natural circulation RHR capability.

2. A shutdown cooling system (SCS), which is a new forced circulation safety class RHR system. This was adopted in addition to the

safety-class core auxiliary cooling system (CACS) and the nonsafety class main loop cooling system (MLCS).

3. Electrically-driven radial-flow main helium circulators.

These design changes significantly upgrade the safety and reliability of the GCFR under normal and accident conditions.

2.2. SCOPE OF REPORT

At the completion of their reviews in 1974, the AEC DOL and the ACRS noted that additional effort and possible design modification would be required to assure adequate core cooling system reliability. The objective of this report is to present a comprehensive set of safety design bases for the GCFR systems used to provide RHR. This report will also describe justification for the selected design bases. This report will present a conceptual design for the GCFR RHR systems and supporting analyses to show that the design bases can be satisfied. The GCFR program for assuring reliable RHR systems will be described in a subsequent document.

2.3. REQUESTED NRC ACTION

The NRC is requested to review this document (PSID Amendment 10, Core Cooling Design Bases) and to concur with the comprehensive set of safety design bases for GCFR RHR systems presented in Section 3. The additional information submitted in the other sections is provided to assist NRC in its evaluation of the design bases. This information includes a conceptual design for the core cooling systems (Section 4), supporting analyses characterizing the safety-related performance of the core cooling systems (Section 5), and a description of the RHR performance verification program (Section 6). Although a dialogue with the NRC concerning the information presented in Sections 4 through 6 is desirable, the NRC is not requested to review these sections for the purpose of concurring in the specific design.

REFERENCES

2-1. "Gas-Cooled Fast Breeder Reactor Preliminary Safety Information Document," General Atomic Report GA-10298, February 15, 1971.

2-2. "Gas-Cooled Fast Breeder Reactor Preliminary Safety Information Document," General Atomic Report GA-10298, February 15, 1975 (Amended 1976).

2-3. "Gas-Cooled Fast Breeder Reactor Preliminary Safety Information Document," General Atomic Report GA-10298, Supplement I, June 8, 1972 (Amended 1974).

2-4. "Gas-Cooled Fast Breeder Reactor - Responses to AEC Questions on the Preliminary Safety Information Document," General Atomic Report GA-10298, Supplement II, October 18, 1973 (Amended 1974).

2-5. "Gas-Cooled Fast Breeder Reactor Preliminary Environmental Report on 300 MW(e) Demonstration Plant," General Atomic Report GA-A12957, May 1, 1974.

2-6. "Preapplication Safety Evaluation of the Gas-Cooled Fast Breeder Reactor," U.S. Atomic Energy Commission Directorate of Licensing, August 1, 1974.

2-7. Stratton, W. R. (Chairman of GCFR Subcommittee of ACRS), Letter to D. L. Ray (USAEC Chairman), November 8, 1974.

3. SAFETY DESIGN BASES FOR GCFR RESIDUAL HEAT REMOVAL SYSTEMS

3.1. INTRODUCTION

The fundamental residual heat removal (RHR) system objective is to adequately âssure that acceptable fuel cladding temperatures and primary system pressure boundary temperatures are maintained for all credible events (or plant conditions) within the design basis which lead to reactor shutdown. The American Nuclear Society (ANS) classification of plant conditions (PCs) has been used to divide all credible GCFR plant conditions into five groups (PC-1 through PC-5) in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. Section 5.1.1 describes the five groups and their expected frequency of occurrence. The basic principle applied in relating design requirements to each of the plant conditions is that the most probable occurrences should be accommodated by the largest design margin and yield the least radiological risk to the public, and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur.

Within this framework, licensing criteria have been developed for use in the design bases of the gas-cooled fast breeder reactor (GCFR) RHR systems. Based upon Nuclear Regulatory Commission (NRC) licensing criteria and positions for the light water reactor (LWR) and the liquid metal fast breeder reactor (LMFBR) and prior review by NRC of the GCFR, the criteria below are believed to meet NRC requirements for RHR. In particular, the principal NRC positions on fast reactor licensing, as defined in the May 6, 1976 letter from R. Denise to the Clinch River Breeder Reactor Project (CRBRP) (Ref. 3-1), are considered to be consistent with the GCFR criteria, which require two safety-class systems for long-term RHR. The GCFR criteria provide a margin of safety in excess of the minimum requirements of the General Design Criteria (GDC) for LWRs (Ref. 3-2), which require one safety-class system for long-term RHR.

In summary, the major requirements of the GCFR criteria are the following:

1. Two redundant safety systems are to be provided for long-term RHR.

   a. Shutdown cooling system (SCS).
   b. Core auxiliary cooling system (CACS).

2. Both the SCS and CACS shall be seismic category I.

3. The SCS and CACS shall be independent from each other.

4. The reliability goal for the RHR systems shall be such that the probability of loss of design core cooling geometry shall be beyond the design basis value.

5. Natural circulation RHR capability shall be adopted with appropriate experimental verification.

The later sections of this report describe how these key requirements are met. Section 4 describes how the SCS and CACS are independent, redundant, seismic category I engineered safety systems both capable of removing all residual heat produced by the core. The reliability goal is expected to be met by the three RHR systems [i.e., main loop cooling system (MLCS), SCS, and CACS]. A detailed analysis of RHR reliability is being conducted as a separate study. Section 6 outlines the key test plan elements to verify the GCFR RHR functions including the natural circulation capability of the upflow GCFR design.

The design described in this report is believed to adequately meet or exceed all the criteria established as the design bases of the RHR systems. The hardware-oriented criteria will be incorporated during the appropriate stages of design.

3.2. GDCs FOR GCFR CORE COOLING SYSTEMS

General Atomic transmitted PSID, Amendment 8, Revision 1, General Design Criteria (Ref. 3-3) to the NRC in July 1979 to obtain their concurrence with recommended changes in the GDC specific to the GCFR. Criterion 34 and Criterion 35, which are applicable to core cooling systems, are quoted below.

3.2.1. Criterion 34: Residual Heat Removal

> " Two independent systems to remove residual heat shall be provided. The safety function of each system shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the primary coolant system boundary are not exceeded. Design techniques that employ diversity in principle shall be used to prevent loss of the safety function. Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for on-site electric power system operation (assuming off-site power is not available) and for off-site electric power system operation (assuming on-site power is not available), the system safety function can be accomplished, assuming a single failure."

The change to two independent RHR systems reflects the present design criteria for the GCFR. This is in conformance with requirements placed on the CRBR plant (Ref. 3-1). Changes are also made to reflect appropriate terminology for gas-cooled reactors.

3.2.2. Criterion 35: Core Auxiliary Cooling System

> "A core auxiliary cooling system shall be provided which has the capability of heat removal at a rate sufficient to prevent any

damage which could interfere with continued effective core cooling assuming a depressurization accident together with a loss of main loop cooling. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for on-site electric power system operation (assuming off-site power is not available) and for off-site electric power system operation (assuming on-site power is not available) the system safety function can be accomplished, assuming a single failure."

The above criteria reflects the design basis for the CACS in the GCFR which is to prevent core damage rather than to perform post-damage heat removal, as in the case of the emergency core cooling system (ECCS).

In addition to complying with the above criteria, the systems must, as a minimum, satisfy the requirements of GDC-36 (Ref. 3-2) for periodic inspection of essential components and GDC-37 (Ref. 3-2) for periodic functional testing of the systems.

3.3. CRITERIA

3.3.1. Safety Core Cooling Function

Transfer of fission product decay heat and other residual heat from the reactor core shall be at a rate such that specified acceptable fuel and cladding design temperatures and the design conditions of the primary coolant system boundary and the internal components are not exceeded.

3.3.2. Reliability Goal

The estimated probability of a loss of design core cooling geometry shall be less than the lower bound of the frequency range for faulted plant condition (PC-5), considering the use of all core cooling and support systems available, including the SCS and the CACS. In other words, all events leading to a loss of design core cooling geometry (even though a coolable core geometry is maintained) shall be classified as beyond design basis plant conditions. The reliability goal shall be met using expected system performance.

3.3.3. Redundancy

Redundancy criteria shall include the following:

1. The SCS and CACS shall each be capable of long-term RHR.

2. The combined availability of all systems, including the SCS and CACS, shall meet the reliability goal starting from normal operation at power.

3. Both the SCS and the CACS shall be capable of cooling the core while maintaining fuel and cladding design limits appropriate to the plant condition, assuming reactor trip and a concurrent single failure.

4. For design basis events of very low probability of occurrence [e.g., the design basis depressurization accident (DBDA)], either the MLCS or the CACS shall provide core cooling such that fuel and cladding or plant damage that would interfere with continued effective core cooling is prevented, assuming a concurrent single failure. (See Section 3.3.20, Note 1.)

3.3.4. Independence

Independence criteria shall include the following:

1. No single event occurring in the SCS shall cause a consequential loss of the CACS safety function, assuming an independent single failure.

2. No single event occurring in the CACS shall cause a consequential loss of the SCS safety function, assuming an independent single failure.

3.3.5. Diversity (See Section 3.3.20, Note 2)

Diversity criteria shall include the following:

1. The CACS shall be diverse from the SCS.

2. Interfaces between the CACS, SCS, and the MLCS shall be subjected to a safety evaluation which includes a failure mode and effects analysis. This evaluation shall demonstrate that the system safety function of the CACS is independent of failures in the SCS or MLCS and that the system safety function of the SCS is independent of failures in the CACS or the nonsafety-related portion of the MLCS.

3. An analysis shall be performed to identify potential points of common mode failure between the SCS and CACS. This analysis should include consideration for credible plant events that could affect both systems, including operator error.

3.3.6. Electric Power

Electric power criteria shall include the following:

1. The SCS and the CACS shall be powered from Class 1E supplies where electric power is required.

2. The SCS and CACS shall be operable on either on-site or off-site power (assuming one is unavailable).

3. The SCS and the CACS shall each be capable of providing adequate cooling following an interruption of its preferred power at any time during an accident sequence which requires core cooling. The single failure criterion shall apply in addition to this interruption. (See Section 3.3.20, Note 3.)

4. The combined capability of the SCS and the CACS shall provide core cooling for at least 2 h following loss of off-site power (LOSP), loss of on-site ac power sources, and a concurrent single failure, subject to the following provison:

   "Where independent, diverse, and redundant Class 1E on-site ac power systems are provided, the loss of only one diverse system need be postulated. (Nonclass 1E electric power systems shall not be assumed to be operable.)"

3.3.7. Leak Detection

Reliable detection shall be provided for leaks at fluid barriers required to maintain the capability of a safety function. (See Section 3.2.20, Note 4.)

3.3.8. Initiation of SCS and CACS Core Cooling

The following initiation criteria shall apply:

1. The SCS and CACS shall each be designed for both manual and automatic initiation.

2. The SCS and CACS shall be automatically initiated in proper sequence when core cooling is not adequate.

3.3.9. Back Pressure

The design of the containment and its isolation system shall provide conditions for adequate core cooling following any credible event, including prestressed concrete reactor vessel (PCRV) depressurization. Credit may be taken for a conservatively calculated back pressure.

3.3.10. System Testing

The SCS and CACS designs shall permit appropriate periodic testing in order to verify:

1. Structural and leak-tight integrity of system components.

2. The operability and performance of active components of the system.

3. The operability of the system as a whole. This may be performed when the plant is shut down.

3.3.11. System Inspection

System inspection criteria shall include the following:

1. The SCS and CACS designs shall provide for appropriate periodic inspection.

2. The designs shall provide at least one of the following: (a) access for in-service inspection of safety-related, heat exchanger tubing to detect tube wall thinning or other defects that could cause heat exchanger failure combined with a postulated DBDA; (b) analysis that demonstrates that such failure is of sufficiently low probability that it need not be considered a design basis event; or (c) analysis that demonstrates that the plant is designed to withstand such a combined failure.

3.3.12. Equipment Classification

The following equipment classification criteria shall apply:

1. SCS and CACS components which are part of the primary coolant system boundary are designated safety class 1.

2. Other safety-related components are designated safety class 2 inside the containment and safety class 3 outside the containment.

3.3.13. Seismic Design Requirements

The following seismic design criteria shall apply:

1. The CACS and SCS shall be included in seismic category I.

2. Loading combinations shall conform with the intent of NRC Regulatory Guide 1.48 (Ref. 3-4).

3. The combined SCS and CACS shall meet the intent of the requirements of NRC Regulatory Guide 1.27, "Ultimate Heat Sink" (Ref. 3-5).

3.3.14. Environmental Design Requirements

The following environmental design criteria shall apply:

1. The environment within the containment due to all design basis events (DBE) or natural occurrence shall not preclude adequate core cooling by the CACS. The same condition shall apply to the SCS, except for some low probability events. (See Section 3.2.20, Note 1.)

2. The SCS and CACS shall be protected from missiles, fluid jets, and/or pressure waves generated by accidents, equipment failures, or natural occurrences.

3. The ultimate heat sink for the SCS is not required to be protected protected from the effects of severe environmental phenomena (other than seismic), provided it can be justified from the site characteristics on a probability basis.

3.3.15. Instrumentation and Controls

The following instrumentation and controls criteria shall apply:

1. Safety-related instrumentation and control (I&C) subsystems associated with the SCS and CACS shall be designed in accordance with the requirements of the Institute of Electrical and Electronic Engineers, Inc. [i.e., IEEE 279 (Ref. 3-6) and IEEE 603 (Ref. 3-7)], and other applicable standards.

2. Setpoints should be established with sufficient margin between the technical specification limits and the set point to allow for (a) instrument inaccuracy, (b) calibration variations, and (c) instrument drift between calibrations. This shall be considered in the safety analysis.

3.3.16. Single Failure Criterion

The single failure criterion shall be defined as follows:

1. Single failure is an occurrence which results in the loss of capability of a component to perform its intended safety functions when called upon. Multiple failures resulting from a single occurrence are considered to be a single failure.

2. The plant protection systems (PPS) and electric power systems associated with the SCS and CACS shall conform to the single failure criteria provided in IEEE 603 and IEEE 379 (Refs. 3-6 and 3-7).

3. Fluid system components associated with the SCS and CACS, including essential support systems, shall be capable of performing their required safety function subsequent to the following single failures:

   a. Active mechanical failure during short-term operation (first 24 h).

   b. Active or passive mechanical failure during long-term operation (after 24 h). (A passive failure is the loss of structural integrity of a fluid-retaining boundary.) Assumption of a single failure is only required in systems which respond to the initiating event by a change in operating state or operating condition. For example, a normally operating SCS could possibly be exempted from the assumed passive failure requirement.

4. When one SCS or CACS loop is unavailable, with the plant operating within Technical Specifications, the single failure criterion does not apply to the remaining loops.

3.3.17. Quality Assurance

Regulatory Guide 10CFR50, Appendix B (Ref. 3-8), applies to the SCS and CACS and essential support systems.

3.3.18. Margin

The following margin criteria shall apply:

1. Sufficient margin shall be available on the time required for SCS SCS and CACS actuation, back pressure requirements, and system

capacity to clearly demonstrate abundance of safety-related core cooling.

2. Adequate margin shall be available in CACS startup time for continued cooling to meet the design temperature limits for the core and the primary coolant components.

3.3.19. Natural Circulation

Experimental verification of the analytical models for natural circulation is required prior to the completion of the construction permit review as part of the system design capability assurance using natural circulation. (See Section 6.)

3.3.20. Notes

The following notes relate to applicable criteria elements:

1. (See Section 3.3.3.) For low probability initiating events, the SCS and CACS need not be redundant between systems to achieve the reliability goal. For example, credit may be taken for main loop coastdown before using the CACS.

2. (See Section 3.3.5.) Two components or systems, having common functional characteristics, are considered to be diverse to the extent that many of the following characteristics are met:

   a. Different physical principles.

   b. Different, independent, motive power sources.

   c. Different manufacturers.

   d. Different instrumentation systems actuated by measurement of different process variables.

   e. Different environmental conditions.

Total diversity in engineered systems is rarely attained. However, a subjective judgment by an experienced designer as to the extent of diversity can be assumed to measure the resistance of diverse systems to common-cause failures. In other words, in the licensing context, arguments for extent of diversity in a particular design are arguments against the credibility of common-cause failures.

NRC statements defining diversity can be found in Refs. 3-9 and 3-10.

3. (See Section 3.3.6.) Loss of preferred power could interrupt the CACS startup sequencer. Such an interruption may require a restart. This process consumes time. It is dependent on sequencer design.

4. (See Section 3.3.7.) This does not imply that automatic dump of the steam generator is required.

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3-2. "General Design Criteria for Nuclear Power Plants," in Code of Federal Regulations, Title 10, Part 50, Appendix A, U.S. Government Printing Office, Washington, D.C., 1977.

3-3. "Gas-Cooled Fast Breeder Reactor Preliminary Safety Information Document," General Atomic Report GA-10298, Amendment 8, Revision 1, July 1979.

3-4. "Design Limits and Loading Combinations for Seismic Category I Fluid Systems Components," Nuclear Regulatory Commission Regulatory Guide 1.48, May 1973.

3-5. "Ultimate Heat Sink," Nuclear Regulatory Commission Regulatory Guide 1.27, March 1978.

3-6. The Institute of Electrical and Electronic Engineers, Inc. (IEEE) Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

3-7. The Institute of Electrical and Electronic Engineers, Inc. (IEEE) Standard 603-1980, "Criteria for Safety Systems for Nuclear Power Generating Stations."

3-8. "Quality Assurance Criteria for Nuclear Power Plant and Fuel Reprocessing Plants," in Code of Federal Regulations, Title 10, Part 50, Appendix B, U.S. Government Printing Office, Washington, D.C., 1977.

3-9. "Auxiliary Feedwater Systems (PWR)," Branch Technical Position APCSB 10-1, Standard Review Plan, Section 10.4, Nuclear Regulatory Commission, November 1975 (NUREG-75/087).

3-10. "Anticipated Transients Without Scram for Water-Cooled Power Reactors," U.S. Atomic Energy Commission Report WASH-1270, September 1973, p. 43.

4. DESCRIPTION OF GCFR DEMONSTRATION PLANT*

This section describes a 350-MW(e) gas-cooled fast-breeder reactor (GCFR) demonstration power plant. It emphasizes the nuclear steam supply system (NSSS), particularly systems related to core cooling, since the remainder of the plant is typical of modern high-temperature steam-turbine practice.

The primary purpose of this plant design is to demonstrate under utility operating conditions the GCFR concept feasibility and the serviceability of fuel and system components to apply to future large commercial nuclear generating stations. A secondary purpose is to demonstrate higher rated fuel use for further system upgrading.

4.1. PLANT DESCRIPTION

4.1.1. General Plant Arrangements

Figures 4-1 and 4-2 show the GCFR demonstration plant arrangement. The primary site structures are the reactor containment and confinement buildings, reactor service building, control and diesel generator building, motor maintenance facility, penetration building, and turbine generator building. The reactor containment building contains the prestressed concrete reactor vessel (PCRV). The PCRV, in turn, contains the reactor core; the helium primary coolant system, comprising the main steam generators and the main helium circulators; and auxiliary heat exchangers and circulators. The containment building is a prestressed concrete structure with a carbon steel inner liner. A reinforced concrete confinement structure surrounds the containment building. A filtered recirculation system with a filtered stack

*Presented to the Nuclear Regulatory Commission (NRC) for information only, not for review and support.

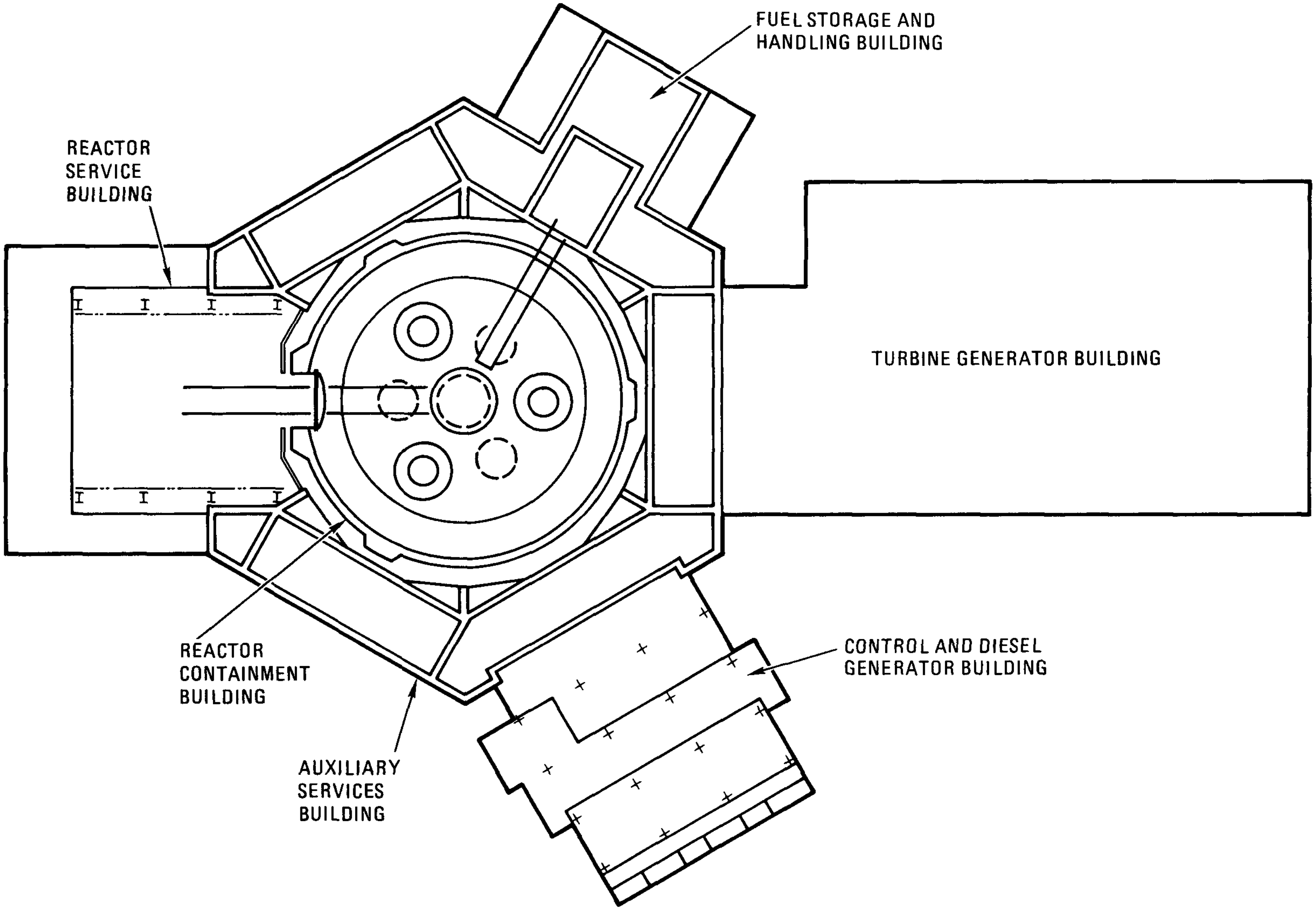

Fig. 4-1. GCFR demonstration plant general arrangement, plan view at operating floor

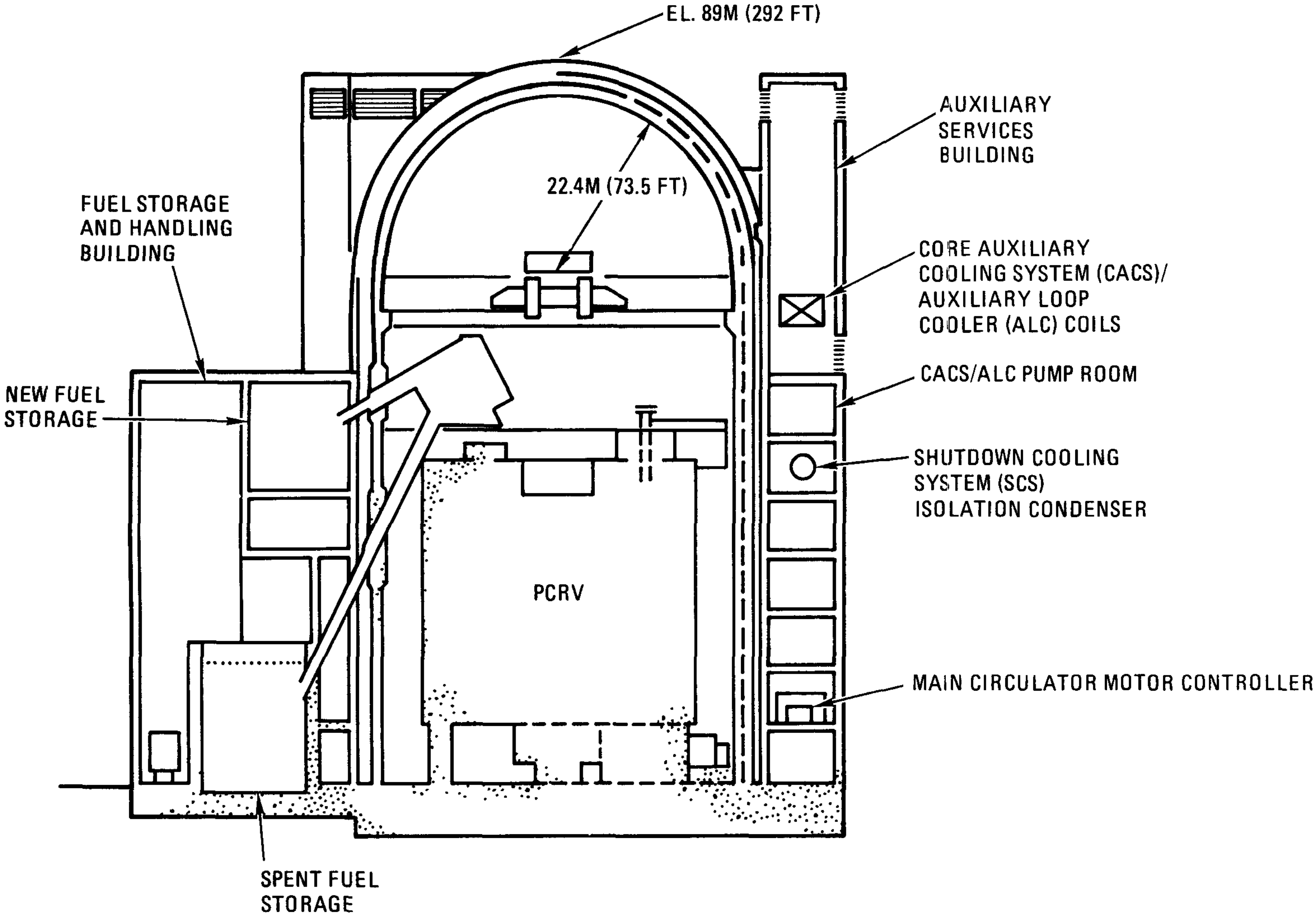

Fig. 4-2. GCFR demonstration plant general arrangement, vertical section

discharge of excess air maintains the interspace between the containment-confinement buildings at subatmospheric pressure. Heat rejection system components are located at an appropriate height adjacent to the confinement building.

Figure 4-3 shows the reactor configuration and its associated primary circuit components within the PCRV. The PCRV, which contains the entire primary system, is reinforced with steel rods and is prestressed after the concrete construction by longitudinal tendons and circumferential wire wrapping. An inner steel liner makes the PCRV leaktight. The PCRV penetrations also have steel liners and steel or concrete closures. Concrete plugs, designed for constant compression, close the major openings. A thermal barrier insulates the liner, and cooling tubes on the concrete side of the steel liners cool the liner and penetrations. The conservative design of this typical PCRV, with its redundant, inspectable and replaceable tension members, precludes a gross failure of the pressure vessel.

The plant reactor coolant system consists of three specific systems:

1. Main loop cooling system (MLCS). The three-loop MLCS transfers heat from the reactor core to the steam generators, producing steam for the plant turbine generator. The remaining two systems area safety systems which ensure that the reactor core will be adequately cooled following reactor shutdown.

2. Shutdown cooling system (SCS). The SCS shares the main circulator, the circulator shaft, and steam generator with the MLCS.

3. Core auxiliary cooling system (CACS). The CACS is a completely separate and independent system having three separate loops.

Section 4.5 describes these cooling systems and their operation more extensively.

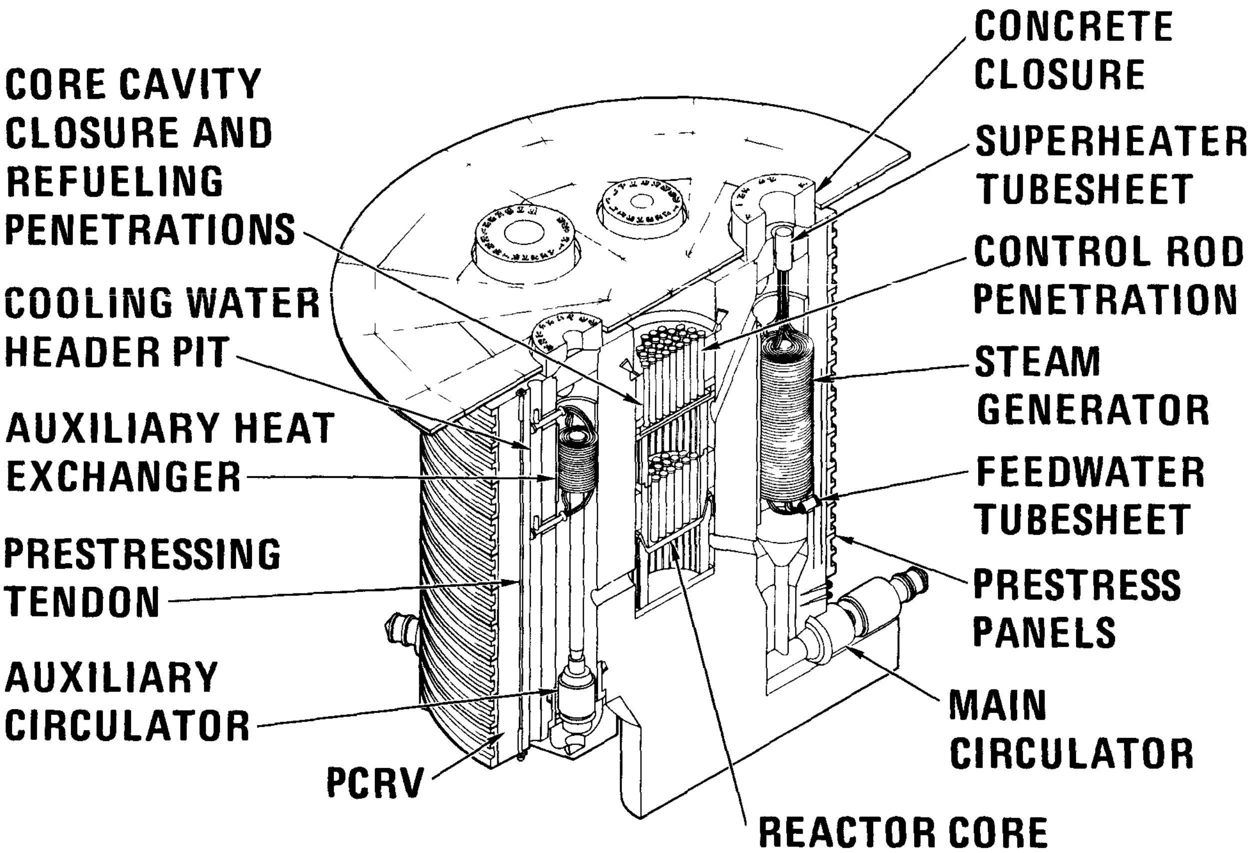

Fig. 4-3. 350-MW(e) GCFR, cutaway of PCRV

4.1.2. Core Elements

The GCFR reactor core is comprised of fuel assemblies, blanket assemblies, control and shutdown assemblies, and radial reflector/shield assemblies. Figure 4-4 illustrates the core general arrangement. A core support grid plate, located in the inlet plenum region, supports the core assemblies. The control rod mechanisms are located in the closures above the core. The hexagonal ducts of the control assemblies fully contain withdrawn control rods. The other assembly ducts incorporate integral exit shielding. The core restraint is a limited free-bow, dual-point lateral restraint design, similar to current U.S. liquid metal fast breeder reactor (LMFBR) designs.

Figure 4-5 illustrates the fuel assembly design. The assembly, a hexagonal duct, houses the individual metal-clad fuel rods, which are spaced by grid spacers. Shielding in the fuel assembly inlet region protects the grid plate. Shielding in the fuel assembly exit assembly shields the reactor exit plenum structures. The coolant flow is orificed so that essentially the same hot spot cladding temperture is reached in each element; orifices are adjusted during refueling.

The GCFR pressure equalization system (PES) vents the fuel rods at primary coolant pressure. Fuel rod pressure is equalized to that of the reactor coolant by collective venting, and the fission gases pass through the vent manifold to the helium purification system. The PES relieves the cladding from mechanical stress caused by external gas coolant and internal fission product gas pressures. This system also limits the release of activity from failed rods to the reactor coolant. It detects and locates fuel elements with failed cladding with activity monitors on the vent lines from separate element groups.

Figure 4-6 is a drawing of the control assembly. The shutdown rods are still being designed. The blanket assemblies are similar to the fuel assemblies, as shown in Fig. 4-7, but the blanket rods are larger diameter and

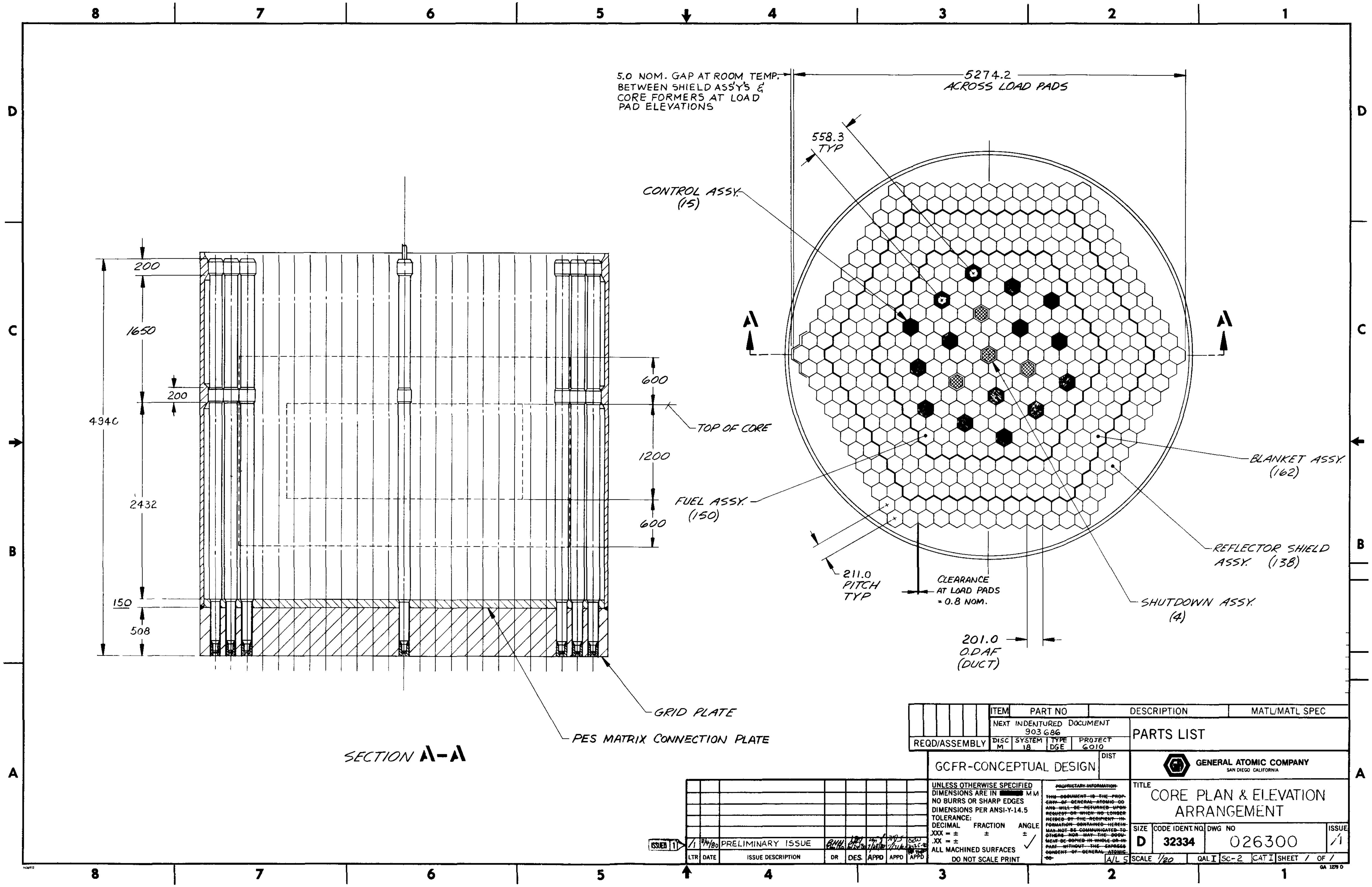

Fig. 4-4. GCFR core plan and elevation arrangement

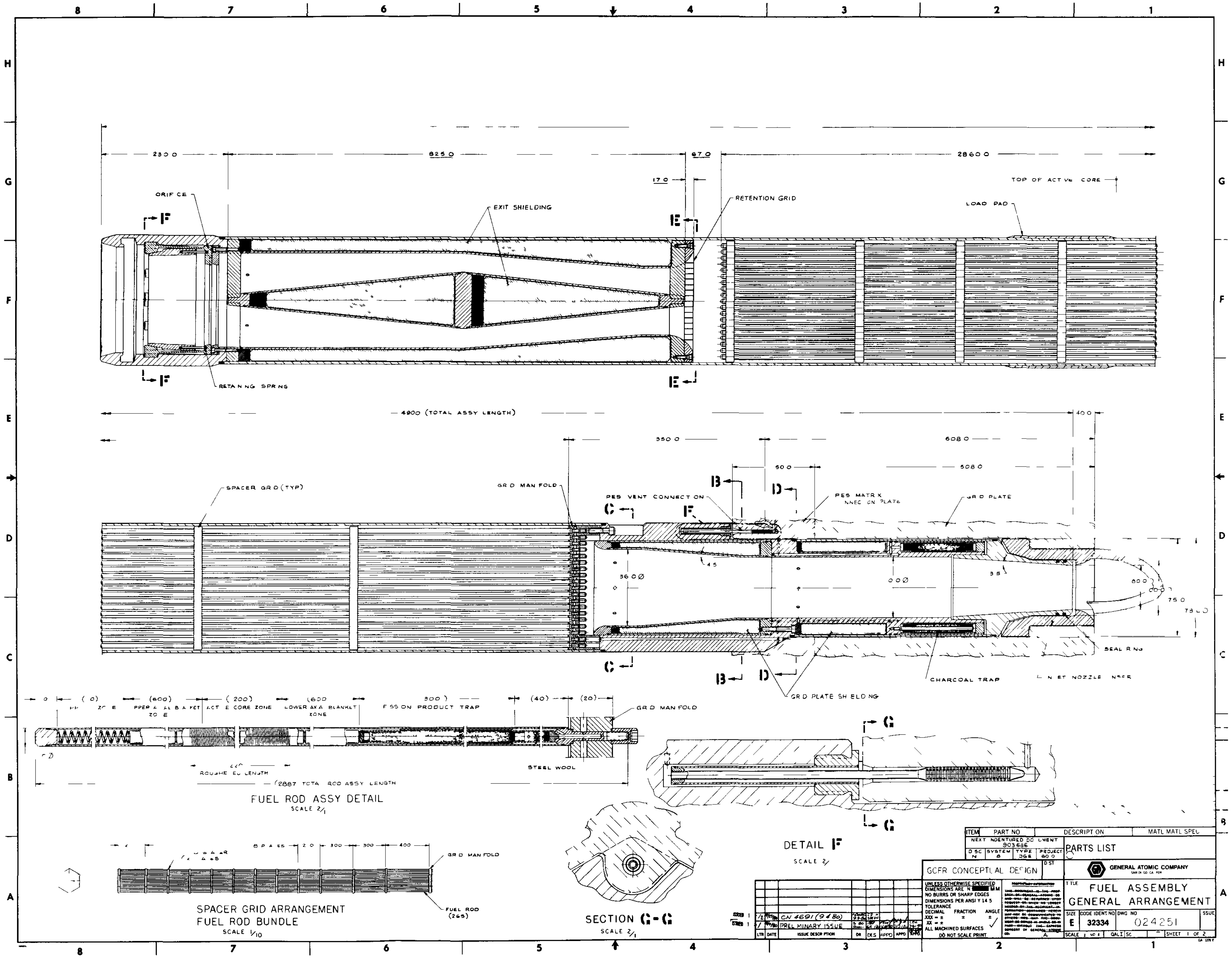

Fig. 4-5. Fuel assembly, general arrangement (sheet 1 of 2)

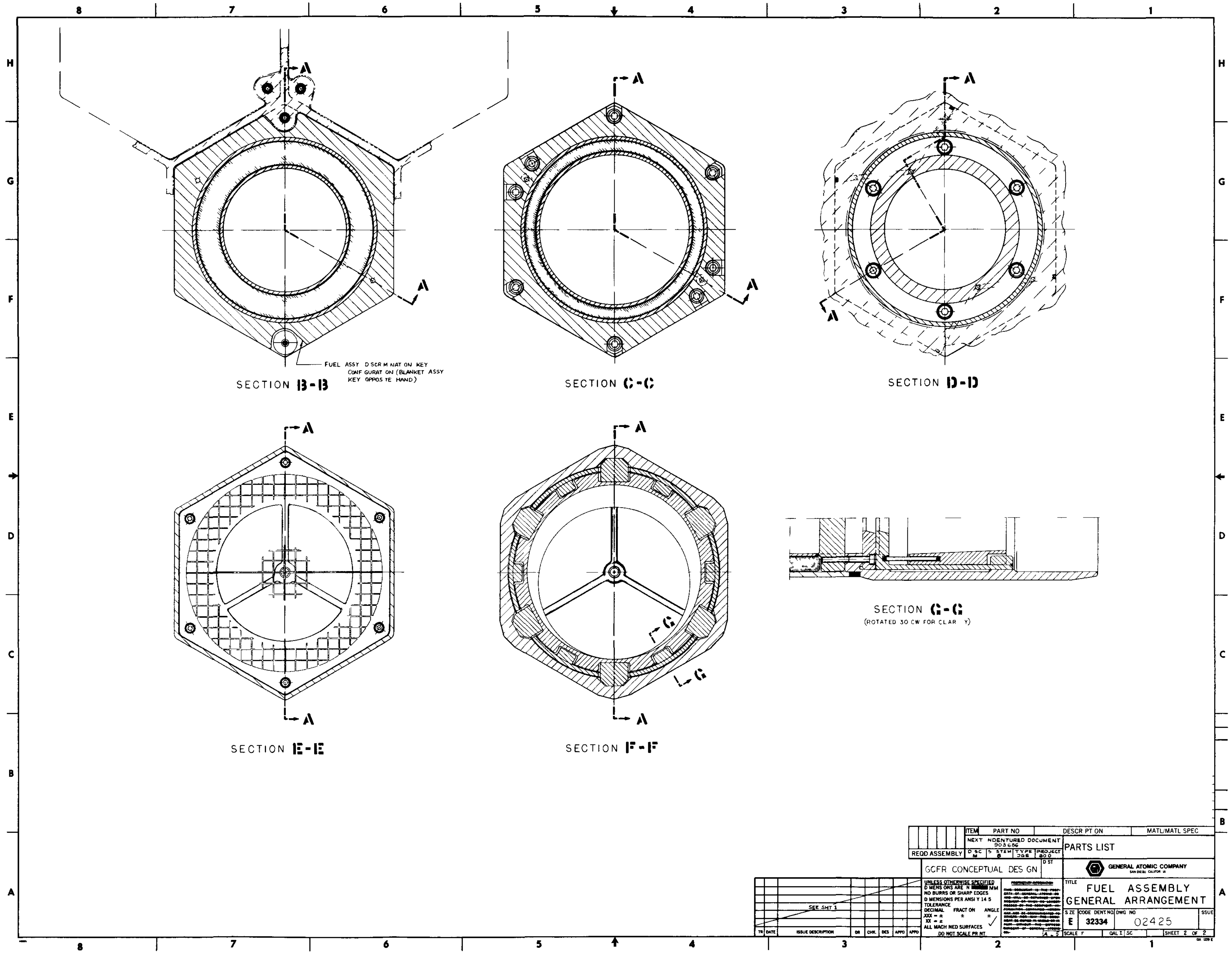

Fig. 4-5. Fuel assembly, general arrangement (sheet 2 of 2)

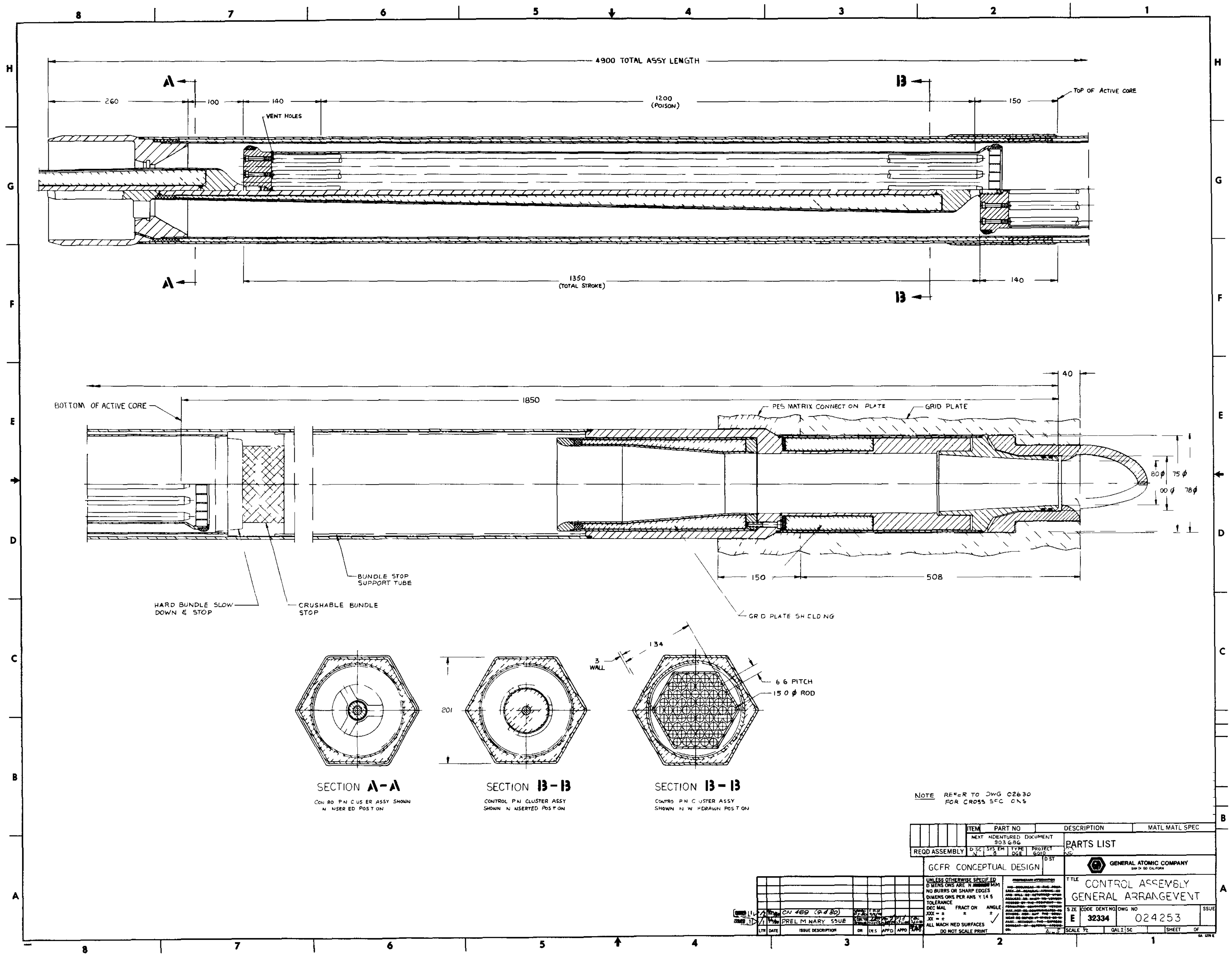

Fig. 4-6. Control assembly, general arrangement

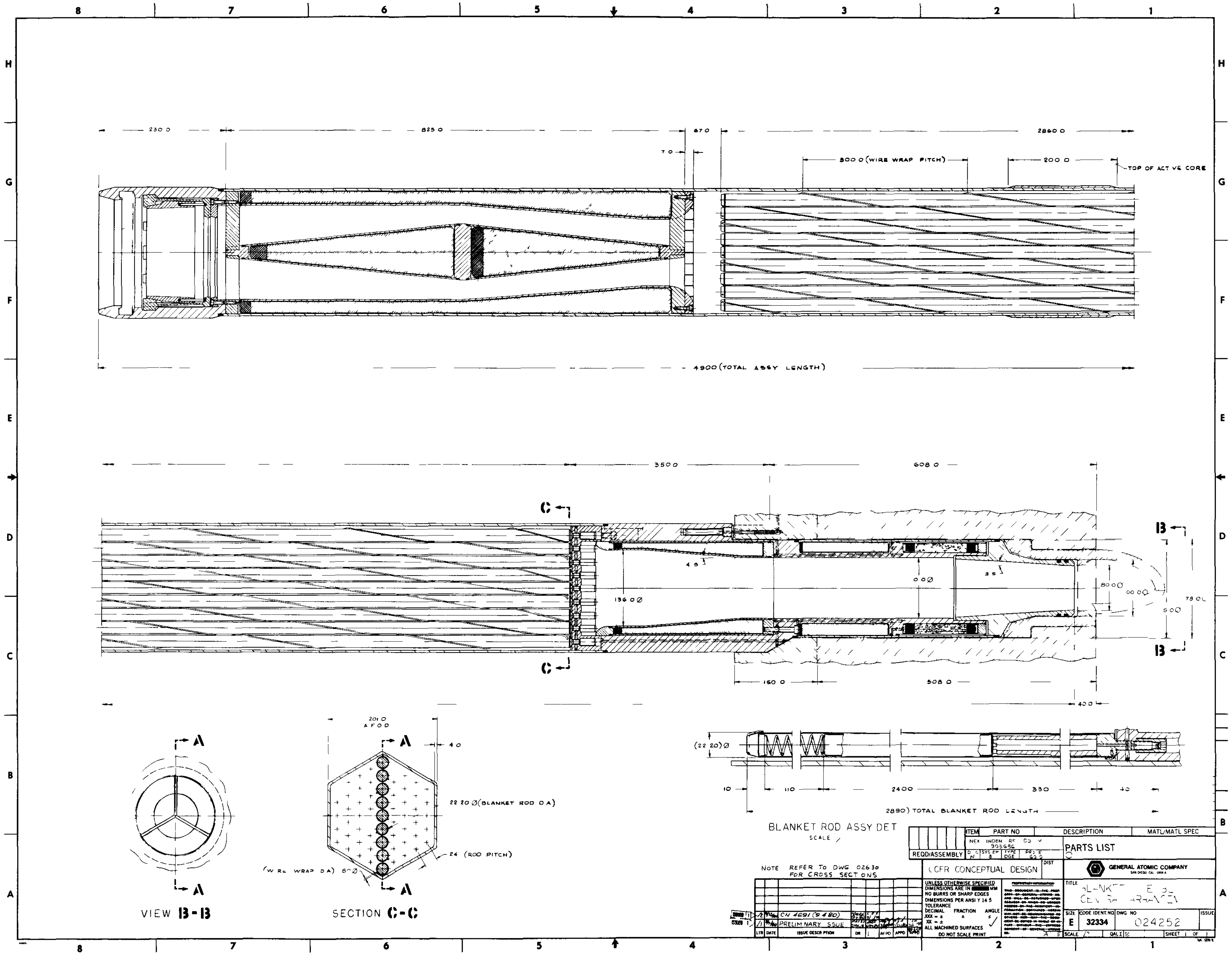

Fig. 4-7. Blanket assembly, general arrangement

are spaced by wire wrap rather than grid spacers. The blanket rods are vented to the PES. The radial reflector assembly consists of a hexagonal steel duct, containing a wire-wrapped bundle of cladded shield material rods. The shield rods are vented directly to the coolant.

Nineteen control assemblies control reactivity. The control rod drives (CRDs) are located above the reactor. Fifteen control assemblies, having an average worth of about $1.80, operate the reactor normally. These rods compensate for burnup and other reactivity effects and can shut down the reactor from any operating condition. The four additional rods, having an average worth of about $3.40, form a backup system capable of independently shutting down the reactor.

Thermal shielding protects the PCRV and the PCRV liner from neutron irradiation. The radial reflector/shield assemblies were described above. The remaining major shielding areas are the upper plenum shielding, radial shield assembly, and lower plenum shielding.

Table 4-1 summarizes the demonstration plant principal design characteristics.

4.2. ENGINEERED SAFETY FEATURES

Engineered safety features (ESFs) are designed to prevent the occurrence or to mitigate the effects of serious plant accidents. The ESFs for the GCFR are the containment system, the residual heat removal (RHR) systems, and the habitability systems.

4.2.1. Containment System

The GCFR containment system provides a boundary against leakage of radioactive material to the surrounding environment for the most serious postulated release of radioactive material.

TABLE 4-1
SUMMARY OF PRINCIPAL DESIGN CHARACTERISTICS

Overall plant performance parameters	
Reactor thermal power	1088 MW(t)
Gross electrical power	(later) MW(e)
Net electrical power output	367 MW(e)
Overall thermal efficiency	33.7%
Breeding ratio	1.31
Doubling time	25.7 yr
Selected NSSS design characteristics	
Reactor vessel system	Multicavity PCRV
PCRV operating pressure	10.5 MPa (1523 psia)
Reactor core system	
Thermal power	1088 MW(t)
Helium flow rate	949 kg/s (2093 lb/s)
Helium inlet temperature	298°C (568°F)
Helium outlet temperature	530°C (986°F)
Helium inlet pressure	10.5 MPa (1523 psia)
Reactor core pressure drop	0.18 MPa (26.5 psi)
Maximum hotspot midwall clad temperature	750°C (1382°F)
Core assembly structural material	Austenitic SS (D9)
Fuel material	$(Pu,U)O_2$
Axial and radial blanket material	Depleted UO_2
Total number of core assemblies	469
Flow control	Orifices variable, adjustable during refueling
Number of fuel assemblies	150
Number of control assemblies	15
Number of shutdown assemblies	4
Number of radial blanket assemblies	162
Number of reflector/shield assemblies	138
Fuel assembly	
Assembly length	4900 mm (193 in.)
Active fuel length	1200 mm (47.2 in.)

TABLE 4-1 (Continued)

Fuel assembly (continued)	
Fraction of active fuel length surface roughened	100%
Upper axial blanket length	600 mm (23.6 in.)
Lower axial blanket length	600 mm (23.6 in.)
Number of fuel rods/assembly	265
Fuel rod spacer type	Spacer grids
Fuel rod o.d.	8 mm (0.315 in.)
Fuel rod pitch	11.5 mm (0.453 in.)
Clad thickness	0.51 mm (0.020 in.)
Pellet o.d.	6.84 mm (0.269 in.)
Radial blanket assembly	
Assembly length	4900 mm (193 in.)
Number of blanket rods/assembly	61
Blanket rod spacer type	Wire wrap
Blanket rod o.d.	22.20 mm (0.874 in.)
Blanket rod pitch	24.1 mm (0.949 in.)
Clad thickness	0.50 mm (0.0197 in.)
Pellet o.d.	21.05 mm (0.829 in.)
Primary cooling system	
Number of loops	3
Main helium circulators	
Compressor type	Centrifugal
Main driver type	ac, synchronous motor
Power	11.2 MW (15,000 hp)
Helium flow rate	316 kg/s (697 lb/s)
Helium inlet pressure	10.27 MPa (1489 psia)
Helium inlet temperature	290.9°C (555.7°F)
Helium pressure rise	0.23 MPa (34.0 psi)
Pony driver type	ac, induction, 1E
Power	313 kW (420 hp)

TABLE 4-1 (Continued)

Steam generator	
Type	Helical coil, axial flow
Heat duty	369 MW
Helium flow rate	309 kg/s (682 lb/s)
Helium inlet pressure	10.31 MPa (1495 psia)
Helium pressure drop	0.04 kPa (6.1 psi)
Helium inlet temperature	520°C (968°F)
Feedwater flow	143 kg/s (315 lb/s)
Feedwater pressure	13.65 MPa (1980 psia)
Feedwater temperature	171°C (340°F)
Superheated steam pressure	10.7 MPa (1550 psia)
Superheated steam temperature	486°C (906°F)
CACS	
Number of loops	3
Auxiliary loop circulator	
Compressor type	Centrifugal
Driver type	Variable speed, induction motor
Core auxiliary heat exchanger (CAHE)	
Type	Helical coil

System design data

	Pressurized Cooldown Natural Convection	Pressurized Cooldown Forced Convection	DBDA
Primary coolant (helium)			
Helium flow per loop [kg/s (lb/s)]	15.0 (33)	29.5 (65)	6.5 (14.3)
Hot helium temp [°C (°F)]	510 (950)	493 (920)	642 (1188)
Cold helium temp [°C (°F)]	298 (569)	333 (632)	249 (480)
Helium pressure [MPa (psia)]	10.10 (1465)	10.44 (1515)	0.225 (32.6)

TABLE 4-1 (Continued)

	Pressurized Cooldown Natural Convection	Pressurized Cooldown Forced Convection	DBDA
Primary coolant (helium) (continued)			
CAHE helium ΔP, [Pa (psid)]	18.6 (0.0027)	76.5 (0.0111)	153 (0.0222)
Molecular weight	4.0	4.0	4.02
Circulator power [kW (hp)]	--	5.59 (7.5)	165 (221)
Heat duty [MW (Btu/h)]	16.5 (5.63×10^7)	24.6 (8.38×10^7)	13.2 (4.53×10^7)
Secondary coolant (H_2O)			
H_2O flow per loop [kg/s (lb/s)]	77.5 (170.8)	252 (555.6)	252 (555.6)
Hot H_2O temp [°C (°F)]	227 (440)	238 (460)	148 (299)
Cold H_2O temp [°C (°F)]	179 (354)	217 (422)	136 (276)
H_2O pressure [MPa (psia)]	9.31 (1350)	9.31 (1350)	9.31 (1350)
CAHE H_2O ΔP [kPa (psid)]	0.17 (0.024)	152 (22)	138 (20)
H_2O pump power [kW (hp)]	--	88.0 (118)	79.8 (107)
Tertiary coolant (air)			
ALC air flow [kg/s (lb/s)]	98.5 (217)	137 (302)	135 (298)
ALC fan power [kW (hp)]	--	119 (159)	117 (157)

<u>Selected balance-of-plant (BOP) design characteristics</u>

Containment building	
Prestressed concrete with carbon steel inner liner	
Free volume	78,282 m^3 (2.76×10^6 ft^3)

Figure 4-8 diagrams the GCFR demonstration plant. The containment building is a seismic category I structure constructed of prestressed concrete and lined with carbon steel. The containment structure is designed to ensure low leakage of radioactive materials and to withstand pressurization to the expected peak pressure following a design basis depressurization accident (DBDA). A reinforced concrete confinement building surrounds the containment structure. The walls of the confinement building are designed for tornado loads. The annular space between the containment and confinement structures collects and confines activity released from the containment and is kept at a slightly negative pressure to limit leakage to the atmosphere. A cleanup system filters all exhaust air prior to atmospheric discharge. The filtration system design ensures that the acceptable upper limit of leakage of radioactive material is not exceeded.

Containment isolation systems close valves in lines penetrating the containment (except lines of safety systems required to operate during accident or shutdown conditions) to ensure that the containment provides the required barrier to release of radioactive gas or particulate matter. Valves may be closed automatically or overridden manually, depending on the type of line penetrating the containment.

4.2.2. RHR Systems

The SCS and the CACS are GCFR RHR systems classified as ESFs. Each system provides an RHR path to an ultimate heat sink. The two systems are independent.

The SCS is a safety-related system designed to provide long-term RHR using forced circulation from the core to the ultimate heat sink. Section 4.5.2. details this system. The SCS is, however, not designed for a group of extremely low probability accidents.

The CACS is a safety-related system designed to provide long-term RHR for all postulated events. The CACS design provides long-term core cooling

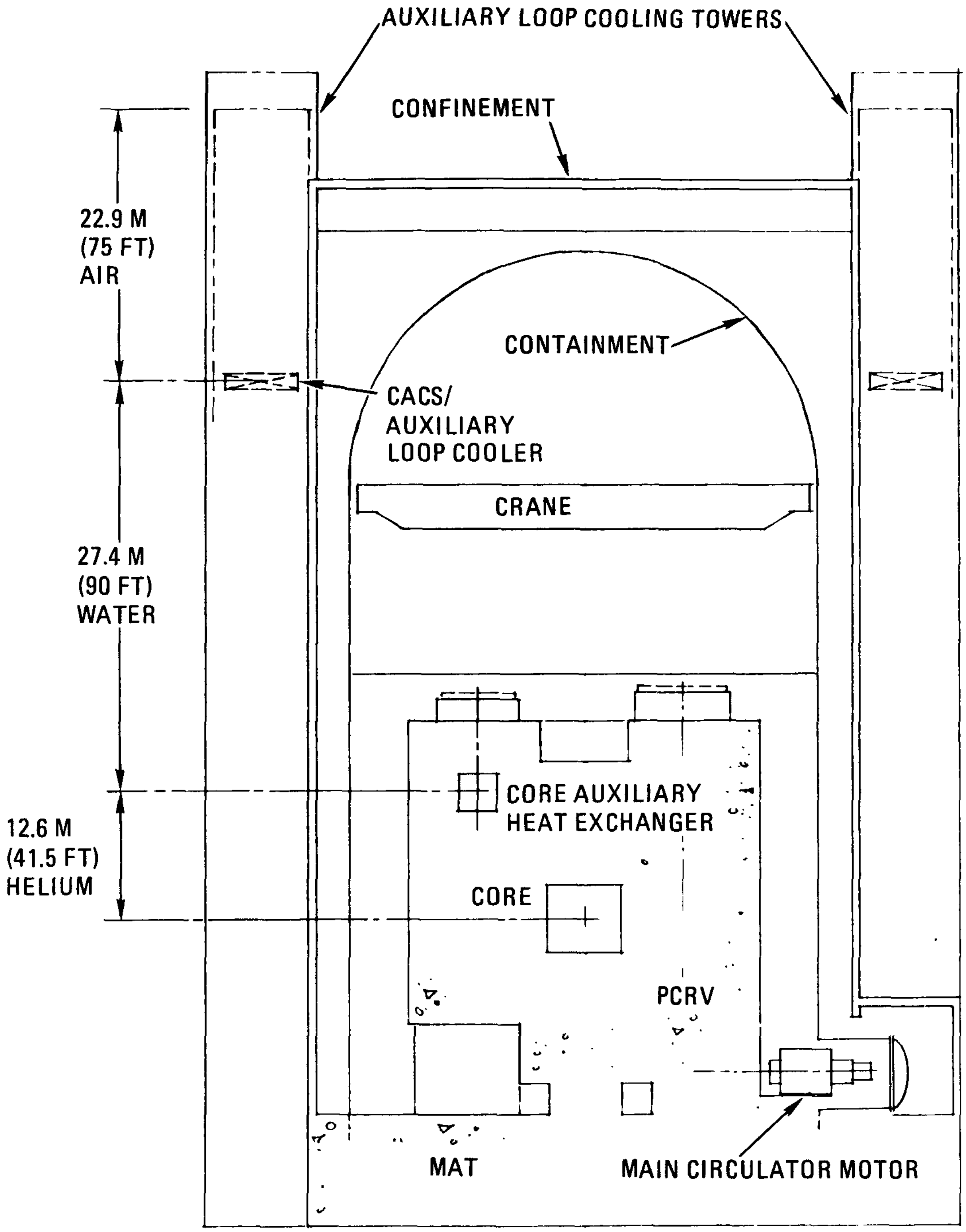

Fig. 4-8. Containment/confinement building

by forced convection to the ultimate heat sink. For pressurized events, the CACS provides a diverse mode of RHR via natural circulation in the primary coolant, secondary (water), and tertiary (air) heat transport loops to the ultimate heat sink. In this mode, the core cooling systems are nearly passive, requiring only minimum equipment operation to ensure adequate core cooling. Sections 4.5.3. and 4.5.4. describe the CACS.

4.2.3. Habitability Systems

The control room habitability system is designed to provide a safe, comfortable, and appropriately equipped location for control personnel during normal and accident operation. The habitability system design features include the following:

1. A low leakage concrete enclosure and specially sealed doors designed against the appropriate thermal loads and activity releases.

2. A heating, ventilation, and air conditioning system, including the required capacity, redundancy, air cleanup and filtration units, pressure control relative to the surroundings, and proper location of intake vents.

4.3. NORMAL CORE COOLING FEATURES

4.3.1. Full Power Operation

4.3.1.1. Design Bases. The principle function of the main loops is to transfer heat from the reactor core to the steam generators and to produce steam for electric power generation. The design basis for power generation is the maximum thermal power condition, or 100% power. However, the main loops must be designed for and withstand all power generating conditions, including load changes and continuous operation at any point down to 25% load.

Tables 4-1 and 4-2 show major design parameters for 100% power. These are nominal (i.e., expected conditions, averaged over the plant life). The plant must be conservatively designed. Some conservatism is required for licensing by statute, but most conservatism stems from sound engineering practices and reduction of commercial and operating risks. This is true for main loop power generation in contrast to CACS or SCS operation, because power generation is not a safety-related function, per normal licensing definition. Both kinds of conservative factors included in the design basis are listed below. Some criteria and designs can be changed as the design progresses, especially those not strictly defined by the NRC and those nonsafety-related aspects, which are primarily under the control of the vendor and the plant owner.

Two major considerations influence safe plant operation during power production: (1) design adequacy and (2) reliability. Design adequacy assures that the plant has sufficient margin and conservatism. Power production reliability (i.e., availability) has been reemphasized because RHR has been adopted as a reliability goal. Thus, if the main loops could be made so reliable that no forced outages occur, then safety systems would not have to be initiated. The discussion below is divided between design adequacy and reliability/availability.

Design Adequacy.

1. Margins were applied in limited areas to assure that the plant can achieve its design power level. The margins are large, because the design is in the conceptual stage, and they are expected to decline as the design is detailed. These margins cover design evolution, uncertainty in predicting performance, and control flexibility. They are applied to core performance (5% in heat transfer film coefficient), helium operating pressure (3 bar), steam generator surface area (5%), and circulator motor power (22%).

TABLE 4-2
HEAT BALANCE AT 1088-MW(t) REACTOR POWER (100%)

	MW
Heat input	
Core	1188.0
Circulator	33.6
Total	1221.6
Heat output	
Steam generator	1106.9
CACS leakages + natural convection loss	9.4
Cold gas heat loss to liner	1.6
Hot gas heat loss to liner and CACS	3.7
Total	1121.6

2. The plant is sized for 100% power, but instrument errors and inaccuracy of plant control must be considered. The plant design must accommodate conditions corresponding to 102% power for the plant design life without component damage and without exceeding long-term fuel cladding temperature limits. (Preliminary analysis indicates instrumentation and controls can limit the power ±2%. This is required to license the plant).

3. Component design should handle extreme operating conditions, including loop-to-loop imbalances, temperature streaks, steam generator tube plugging, uncertainties, and similar departures from nominal conditions. Combinations of extreme conditions should be considered. (This is required to license the plant.)

4. The plant should be designed for a 30-yr life at an average 80% capacity factor. The plant is designed to be capable of continuous operation under fully automatic, semi-automatic, and manual control at any power in the operating range. The plant is base loaded and is not designed for load following, but has load changing capability within its operating range at rates of load change up to those shown in Table 4-3 and for the number of cycles at various rates and over various ranges given in Table 4-4.

Reliability/Availability.

1. The plant should be designed for 90% on-line availability (not including initial rise-to-power and special testing).

2. The plant should have the ability to accept a trip of a single loop while maintaining operation and to continue operation at reduced load with one or two main loops out of service [so long as adequate SCS capability of the shutdown loop(s) is maintained].

   The plant should also be capable of rejecting up to full load from the distribution network, to operate supporting its own auxiliary

TABLE 4-3
DESIGN RATES OF ELECTRICAL LOAD CHANGE

Maximum rate of load change (for changes > 10%)	3%/min
Maximum step load change	10%
Total time for step load changes	(later)
Minimum time between step load changes	(later)

TABLE 4-4
NORMAL PLANT TRANSIENTS

	Design Number of Occurrences
Startup from refueling conditions	140
Startup with full helium inventory	517
Shutdown to refueling conditions	81
Shutdown with full helium inventory	81
Rapid load increase (3%/min maximum) (25% → 100%)	1500
Rapid load decrease (3%/min maximum) (25% → 100%)	1500
Step load increase (+10%)	900
Step load decrease (-10%)	900

loads, to operate with the turbine-generator tripped and steam bypassing the turbine, and to subsequently restart the turbine.

Finally, the plant should be capable of continued operation with one CRD unit out of service.

3. Balance-of-plant (BOP) features shall be considered to enhance the redundancy and independence of the main loop NSS. Redundant feedwater heaters, pumps, and steam systems should be provided, as appropriate, such that loss of individual BOP components would not necessarily shut down the plant and preclude electric power generation.

4.3.1.2. System Description. The MLCS consists of three independent and separate helium loops, with associated water/steam piping arranged outside the PCRV. The helium loops are connected to the reactor cavity by upper and lower cross ducts within the PCRV.

Each helium loop contains a steam generator, a main helium circulator, and a loop isolation valve. The loop components are contained in separate PCRV cavities and are accessible through PCRV top and bottom penetrations. Each steam generator cavity closure is a composite of steel and reinforced concrete.

The steam generator consists of economizer, evaporator, and superheater sections and is a once-through unit with a helically-wound tube bundle with upflow boiling. Hot helium from the core flows from the upper reactor cavity via the cross ducts into the steam generator cavities. The helium passes downward across the superheater, evaporator, and economizer sections, flows into the associated main circulator inlet plenum, then is compressed by the circulator to 0.23 MPa (34 psi) above the circulator inlet pressure. Helium then leaves the diffuser, passes through the main loop isolation valve (MLIV), and enters the reactor inlet plenum via the lower cross ducts. Figure 4-9 shows the helium flow path through a single MLCS loop.

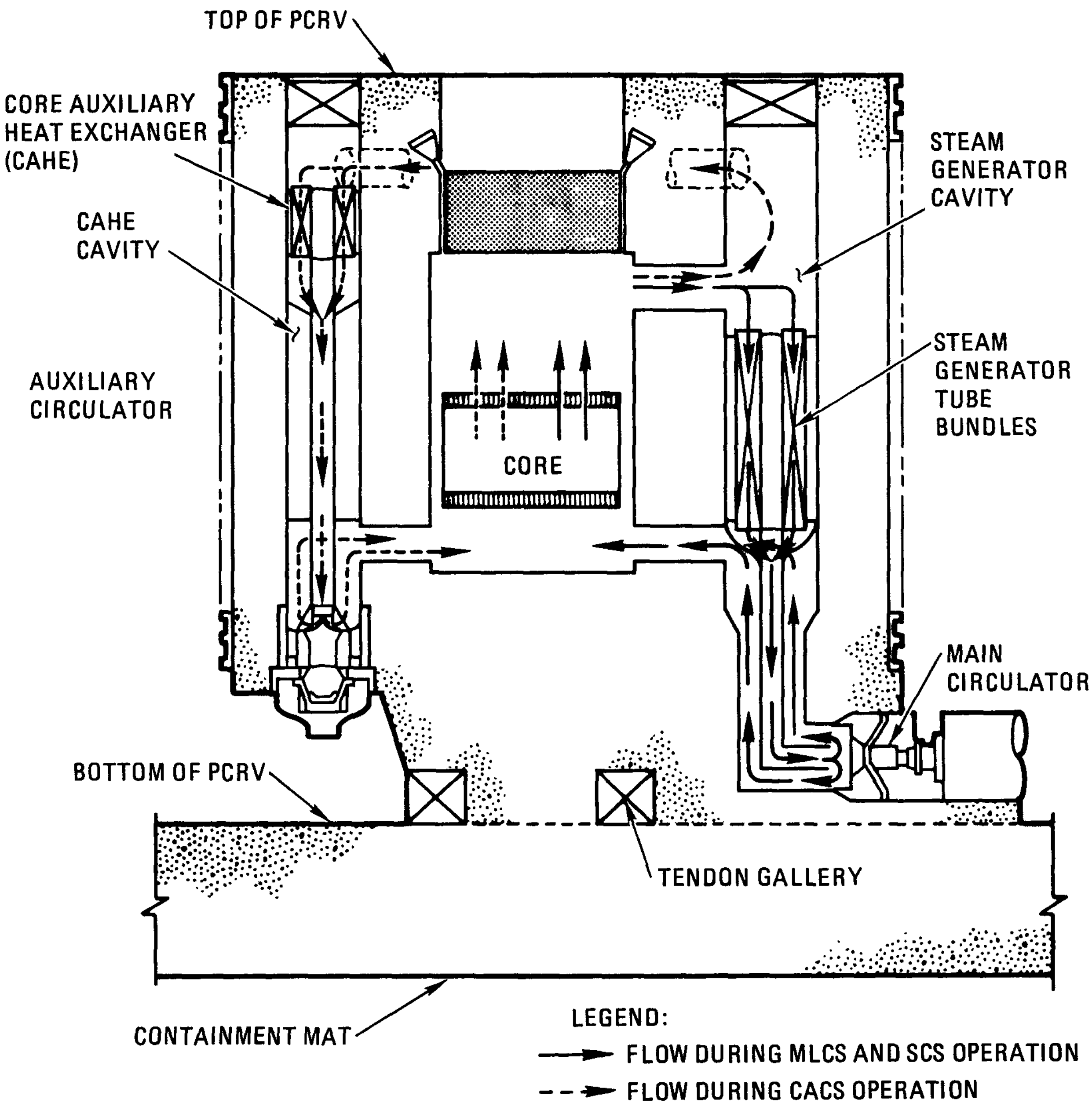

Fig. 4-9. GCFR three-loop NSSS, primary coolant flow arrangement

The main circulators maintain main coolant flow at a flow rate, temperature, and pressure consistent with the reactor core and the steam generator performance requirements. Each main circulator consists of a radial compressor driven by a variable speed synchronous motor external to the PCRV and connected to the compressor shaft through a solid coupling.

A MLIV is downstream of the circulator diffuser. The GCFR demonstration plant has a butterfly reference design isolation valve which self-opens by means of the gas flow pressure differential. Under normal operating conditions, the valve is open. When a main circulator is shut down, the valve in that loop is gravity-actuated to a closed position. A fail-safe actuator is available to close the valve should gravity actuation fail.

The water/steam piping and associated equipment outside the PCRV extremity are part of the BOP. Superheated steam from the steam generators is transported to the main steam turbine. After expansion, the wet steam flows into the main condenser. Condensate pumps then deliver the water through the low pressure feedwater heaters to a deaerator. From there, steam turbine-driven main boiler feedwater pumps return the water through high pressure feedwater heaters to the steam generators. The dual main condenser is cooled by the circulating water system, which has two circulating water pumps and lines, but a single main cooling tower.

Figure 4-10 shows the MLCS flow diagram during normal plant operation.

4.3.1.3. Operation and Control. The MLCS is designed to function during all normal plant operating conditions and the more frequently expected failure conditions. These conditions include plant startup, normal station power production, operator-controlled station shutdown, rapid power runbacks (reactor alone or reactor and the main turbine), single secondary loop shutdowns and trips, turbine generator trips, loss of off-site power (LOSP) (with rapid runback to household power levels), reactor trip, and all other accident conditions, including a design basis depressurization accident (DBDA). Normal station shutdown and reactor trip conditions bring on the MLCS RHR operating mode, which is detailed in Section 4.5.1.

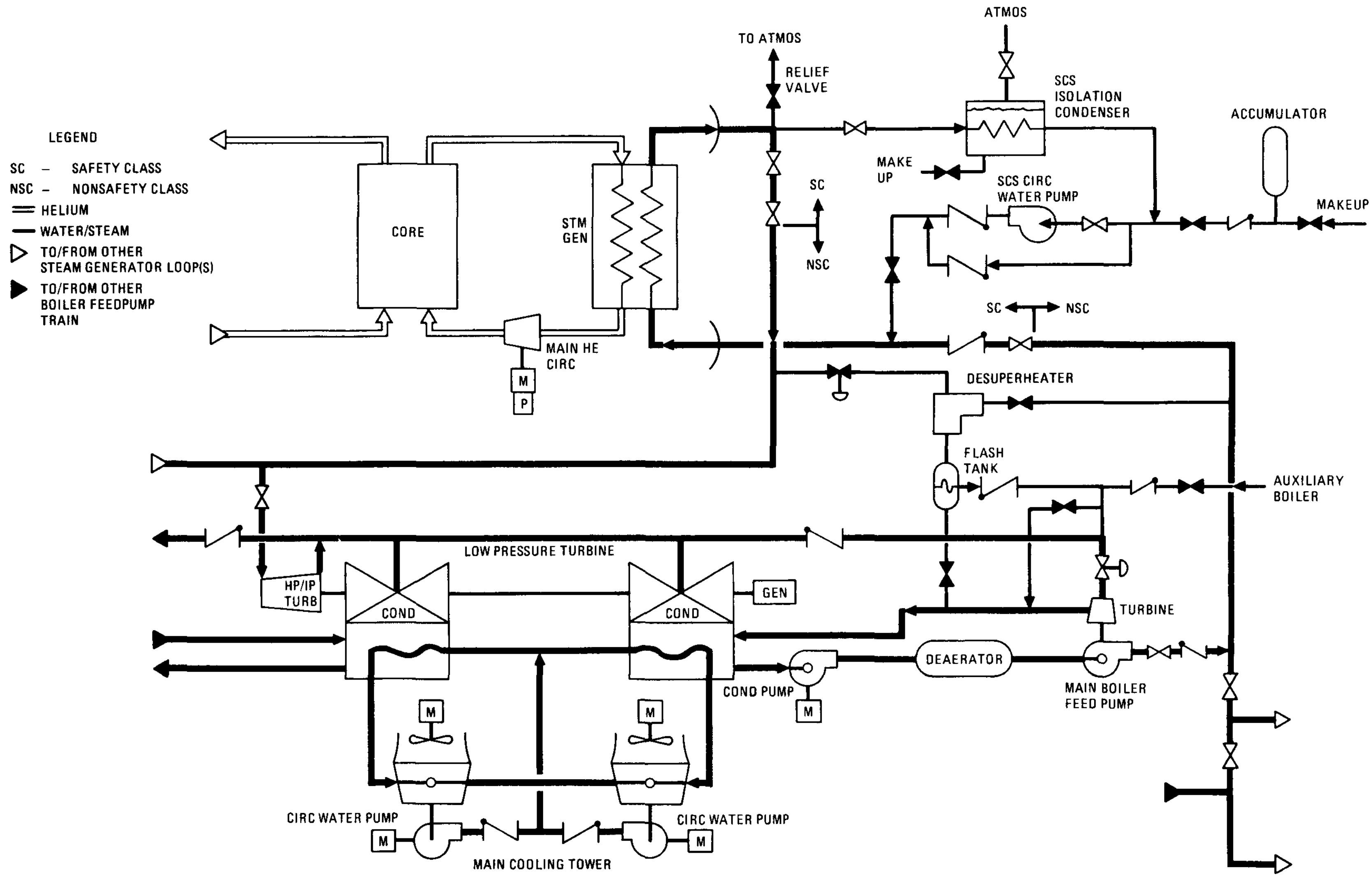

Fig. 4-10. GCFR main and shutdown cooling systems, normal MLCS plant operation, process flow

The normal plant control system reaction to a particular accident condition is important. The plant control system has the following objectives:

1. Maintain preset main steam temperature and pressure.

2. Regulate reactor power relative to main turbine load (reactor follow turbine).

3. Balance steam generator load.

The system satisfies these objectives by the following actions:

1. Using the reactor rods to control reactor power and, consequently, main steam temperature.

2. Adjusting the boiler feed pump turbine valve to control feedwater flow and, consequently, main steam pressure.

3. Varying the speed of the helium circulator motors to maintain helium flow proportional to feedwater flow and to maintain each of the three steam generators at the same exit steam temperature, balancing the thermal loads.

Figure 4-11 shows this multiloop system structure. Figure 4-12 gives the specific control system configuration, showing only one of the three plant secondary loops.

The main steam temperature at the steam generator exit is controlled throughout the normal load range by adjusting reactor power. This is accomplished by measuring the loop average steam temperature, conditioning the signal, and generating a neutron flux demand signal. The flux controller then adjusts the position of the control rods to vary reactor power. Limits

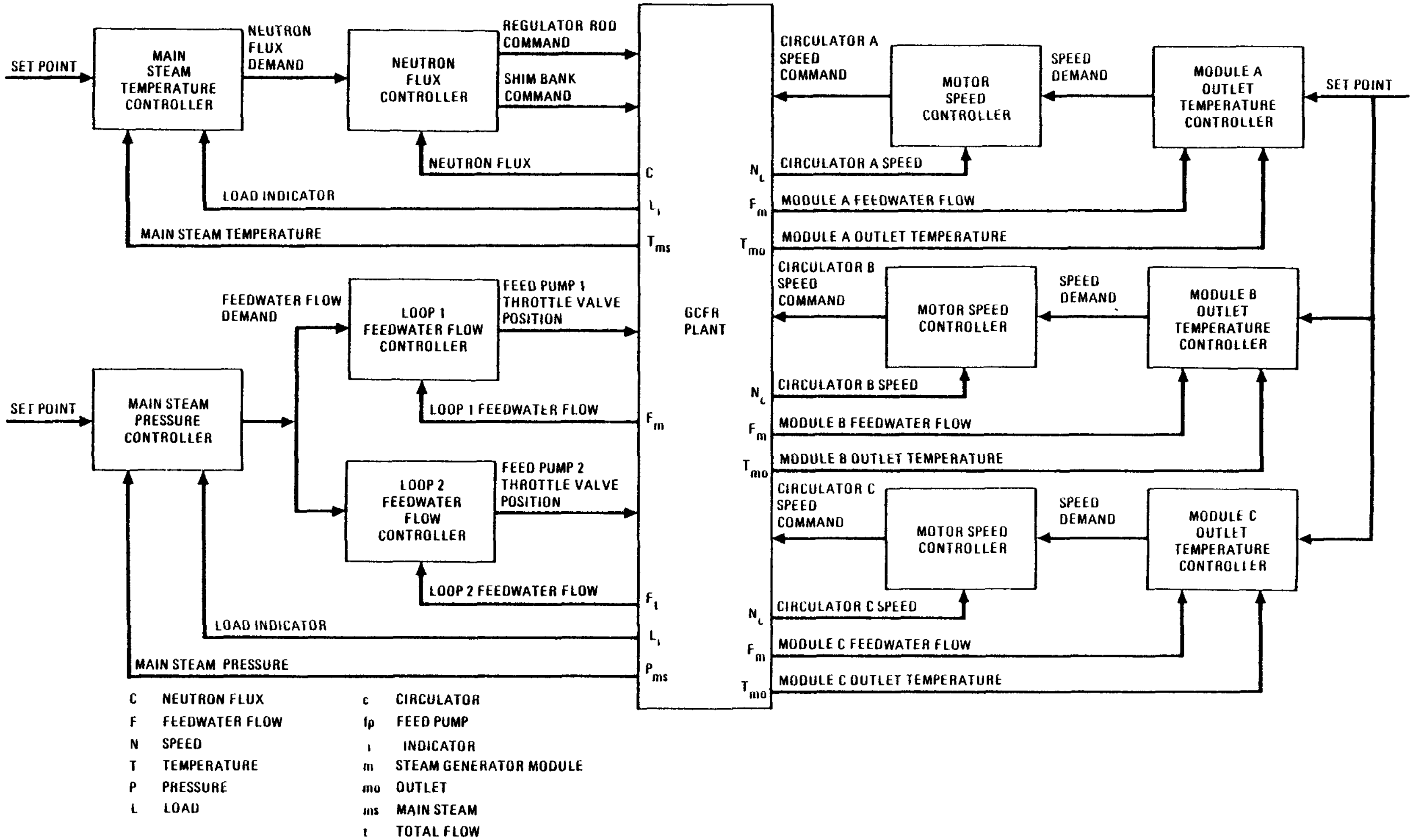

Fig. 4-11. Control system structure

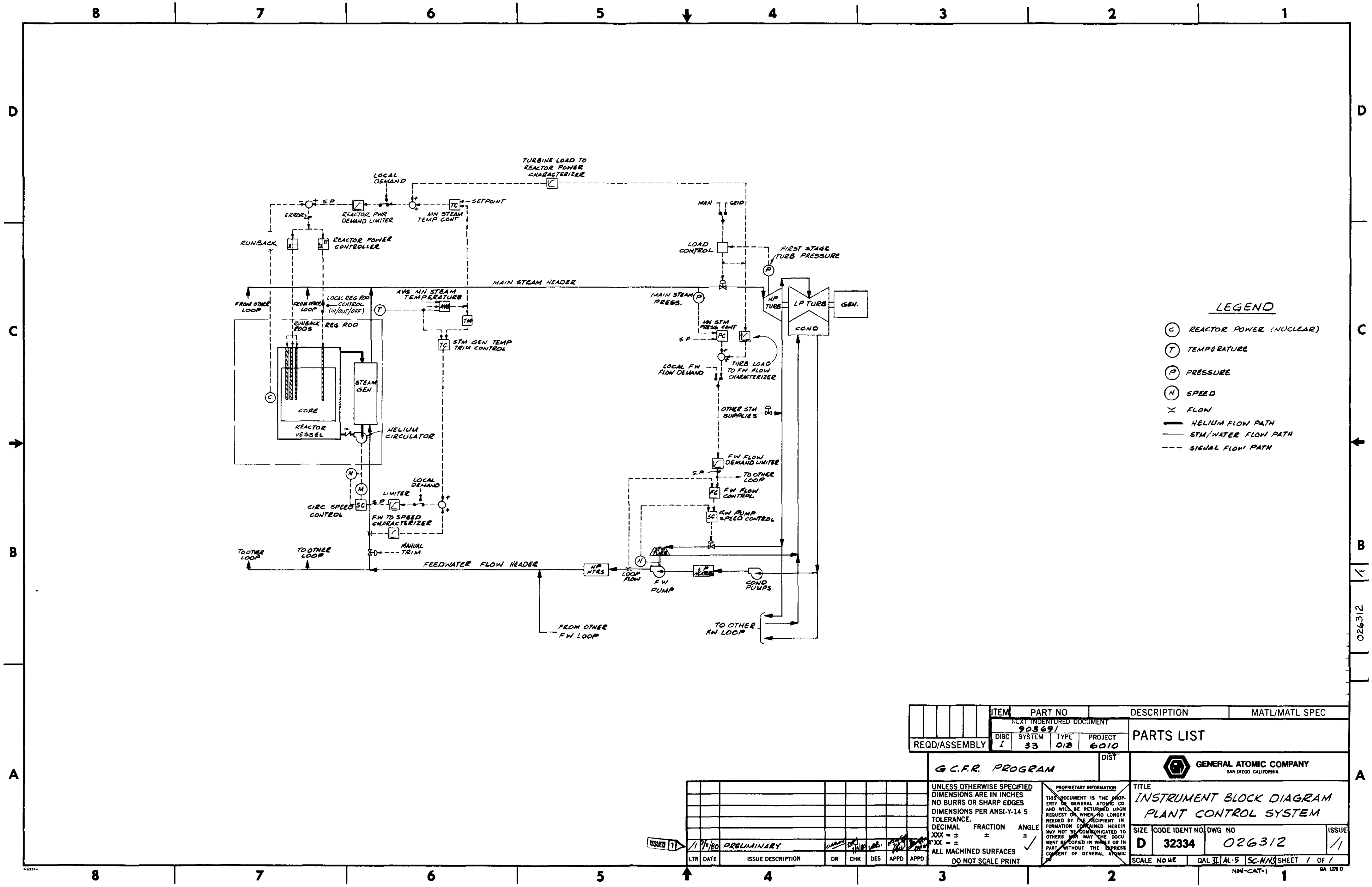

Fig. 4-12. Normal operating plant control system configuration

are imposed on flux demand variation to prevent excessive flux and reactor power excursions.

The steam pressure at the inlet of the high-pressure turbine stop valves is controlled throughout the normal load range by manipulating feedwater flow. A feedwater flow demand is generated as a function of the error between the measured steam pressure and the pressure setpoint value. The feedwater flow controller then compares the feedwater flow demand with measured feedwater flow and maintains the flow at its demanded value. The compensated feedwater flow error signal controls the position of the feed pump turbine valve which, in turn, varies the speed in each steam-driven feed pump to produce the required feedwater flow.

Both the neutron flux controller and the feedwater flow controller have a second input signal. This is a load signal derived from the measured high-pressure turbine first-stage pressure. For a change in load, this feed-forward signal adjusts the neutron flux and feedwater flow rate in anticipation of a change in main steam temperature and pressure. By providing a feed-forward signal to these controllers, the necessary process changes required by plant load changes are begun at the time of the load change, instead of waiting for the buildup of process error measurements. This reduces process transients, especially in large or rapid load changes.

The circulator speed demand signal contains two components. One is a functional relationship designed to maintain helium flow through each steam generator in a fixed proportion to the feedwater flow through that steam generator for the normal plant load range. The other is based on a setpoint computed to be the average of the three measured steam generator outlet steam temperatures. This average temperature is compared with the actual outlet temperature in a particular loop to obtain the temperature error signal for that loop. This temperature error signal acts as a trim function on the helium flow to maintain the steam temperature at the outlet of each steam generator module near the average outlet temperature for the three modules.

Circulator speed is regulated by a closed-loop motor speed controller. Both the amplitude and frequency of the voltage applied to the motor are varied to control motor speed.

4.3.2. Refueling Operation

4.3.2.1. Design Bases. Table 4-5 shows the refueling conditions using the main and CACS loops. During refueling, the core outlet plenum temperature must be maintained below 288°C (550°F). Refueling does not establish the size nor limit the design of either the MLCS or CACS loops. In fact, two of the three main loops are the design basis for normal refueling, but one is adequate, as shown in Table 4-6. The main loops must, of course, be able to operate in this mode, which requires controls and PPS bypasses down to low power, low flow rate, etc. The design will permit use of only one main loop, whenever appropriate, to perform refueling.

The CACS has the same refueling requirements as the MLCS. Two of the three CACS loops are the design basis for normal refueling, but one of three is adequate, as shown in Table 4-6.

4.3.2.2. System Description. Refueling operations are predicted on a 3-yr core life, whereby one-third of the reactor core is replaced each year with new fuel. All reactor core assemblies are transferred through refueling penetrations in the core cavity closure of the PCRV. These penetrations contain either control rod drives (CRDs) or instrument trees which must be removed from the penetration prior to installing the fuel handling machine.

Helium coolant, supplied via the fuel handling equipment, will flow through each fuel assembly from the moment the assembly is disengaged from the grid plate to the time the assembly can be water cooled in the spent fuel chute or the storage pool. The flow will be adequate to maintain the assembly cladding temperature at <315°C (600°F).

TABLE 4-5
EXPECTED NSS OPERATING PERFORMANCE, REFUELING, TWO DAYS AFTER SHUTDOWN,
GCFR DEMONSTRATION PLANT

	Two MLCS Loops	Two CACS Loops
NSS summary		
Reactor thermal power [MW(t)]	5.46	5.46
(%)	0.5	0.5
Primary coolant system		
Number of loops operating	2	2
Total helium flow rate [kg/s (lb/h x 10^6)]	50.1 (0.082)	67.2 (0.110)
System helium pressure at circulator outlet [MPa (psia)]	0.0931 (13.5)	0.0931 (13.5)
System helium pressure drop [MPa (psi)]	2.34 (0.34)	1.38 (0.20)
Circulator inlet temperature [°C (°F)]	115.6 (240)	115.6 (240)
Core inlet temperature [°C (°F)]	120.3 (248.5)	118.6 (245.5)
Core outlet temperature [°C (°F)]	221.6 (430.9)	193.6 (380.5)
Steam generator inlet temperature [°C (°F)]	221.6 (430.9)	193.6 (380.5)
Helium circulator (per loop)		(Aux. Circ.)
Helium flow rate [kg/s (lb/h x 10^6)]	25.1 (0.041)	33.6 (0.055)
Power input to circulator (MW)	0.17	0.22
Steam generator (per loop)		(CAHE)
Helium flow rate [kg/s (lb/h x 10^6)]	25.1 (0.041)	33.6 (0.055)
Feedwater flow rate [kg/s (lb/h x 10^6)]	141.7 (0.232)	(later)
Feedwater inlet temperature [°C (°F)]	115.6 (240.0)	115.6 (240.0)
Feedwater inlet pressure [MPa (psia)]	0.79 (115)	(later)
Superheater exit temperature [°C (°F)]	160.3 (320.5)	(later)
Superheater exit pressure [MPa (psia)]	0.62 (90)	(later)

TABLE 4-6
SHUTDOWN DEPRESSURIZED COOLING CAPABILITY
(MINIMUM TWO DAYS AFTER SHUTDOWN)

	Reactor Power (MW)	Helium Flow [kg/s (lb/h x 10^6)]	Core Outlet Temperature [°C (°F)][(a)]
1 MLCS loop	5.46	33 (0.054)	275 (527)
1 CACS loop	5.46	33.6 (0.055)	269 (516)

[(a)]Refueling permitted for core outlet temperature <288°C (550°F).

Core cooling during refueling can be provided by the MLCS, SCS, or CACS. Use of the two MLCS loops or two CACS loops is the design basis requirement (see Sections 4.5.1.1 and 4.5.3.1). However, the design will allow using one MLCS loop, one CACS loop, or the necessary number of SCS loops, whenever appropriate. Operating with fewer than the design basis requirements makes necessary such items as control and/or PPS bypass conditions not yet considered.

Sections 4.5.1, 4.5.2, and 4.5.3 describe the MLCS, SCS, and CACS, respectively.

4.3.2.3. Operation and Control. If refueling is carried out with MLCS cooling, the MLCS will operate in the long-term RHR mode with steam from auxiliary boilers driving the main boiler feedpump(s) and miscellaneous steam users. The refueling core cooling operation will be similar to that described in Section 4.5.1.

If refueling is carried out with CACS cooling, cooling loop(s) operation will be similar to that described in Section 4.5.3.

Refueling would not normally be done with the SCS. However, refueling cooling capability exists with SCS loops. If refueling with SCS cooling were appropriate, operation would be similar to the long-term operation described in Section 4.5.2.

4.4. REACTOR TRIP SYSTEM

The GCFR PPS contains two diverse and redundant reactor trip systems, the primary and secondary trip systems. Each trip system has an independent and diverse logic system. Additionally, the PPS contains the RHR initiation and termination systems (described in Sections 4.5.1.3, 4.5.2.3, and 4.5.3.3). The following sections summarize the design of the reactor trip systems used in the analysis of the core cooling system performance.

4.4.1. Primary Reactor Trip System

The primary trip system releases the gravity-actuated control rods (see Sections 5.1.3.5 and 5.1.5), using the signals from the primary trip parameters (Fig. 4-13). A reactor trip signal from the primary trip system opens five trip breakers (connected in a two-out-of-three trip matrix), interrupting power to the CRDs. The loss of power to the trip magnet coils causes the CRDs to release the rod control assemblies, which then fall by gravity into the core. The total negative reactivity worth available for primary reactor trip at the beginning-of-life (BOL) is ~\$15.66. Additionally, a buffered signal is sent to the plant control system to trip the main turbine and initiate primary and secondary coolant flow reduction for cooling on the MLCS.

4.4.2. Secondary Reactor Trip System

The secondary trip system releases the shutdown rods (see Sections 5.1.3.5 and 5.1.5) using signals from the secondary trip parameters (Fig. 4-14). A reactor trip signal from the secondary trip system initiates removal of the holding current from the torque motor, allowing the drive line and control rod to fall by gravity. A kickoff spring acts on the drive line to overcome system inertia and accelerate motion. The rate of fall is maintained at a velocity consistent with the required rod insertion time by a resistance connected across the motor windings. The motor then acts as a generator loaded by a fixed resistance.

Total insertion time is ~10 s. To assure that all rods have been fully inserted, after a 15 s delay, the shutdown rod drive motors are energized, driving any potentially stuck rods fully in. The total negative reactivity worth available for secondary reactor trip at BOL is \$13.60.

Additionally, a tertiary triggering mechanism to be provided for only the shutdown rods will be automatic, self-actuating, and independent from the PPS. Several concepts have been considered for this mechanism, but design selection has not been made.

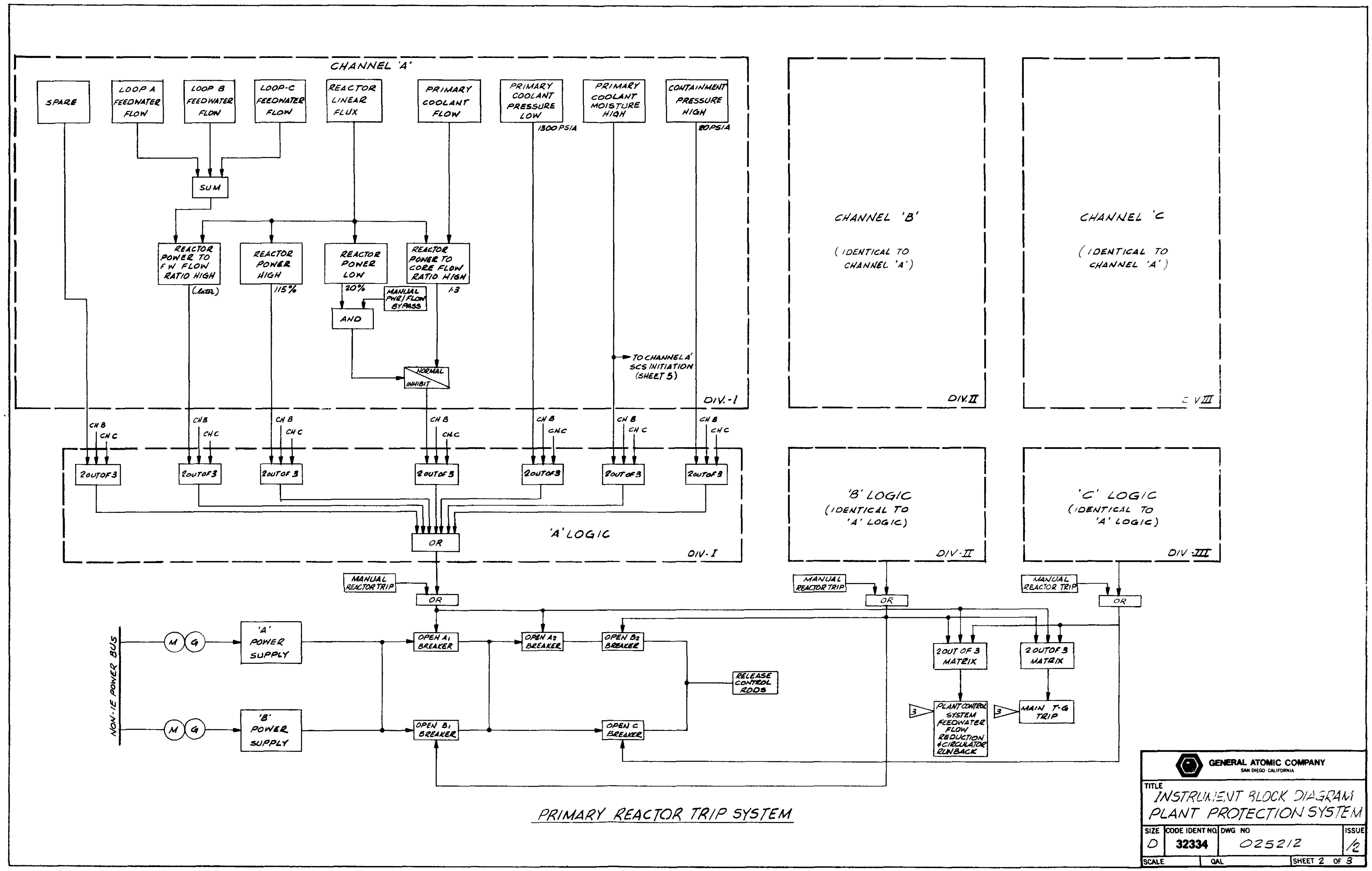

Fig. 4-13. Primary reactor trip system

4.5. SHUTDOWN CORE COOLING FEATURES, OVERVIEW OF MLCS, SCS, CACS, AND NATURAL CIRCULATION CACS

Reactor cooling is provided by three systems: (1) the MLCS, (2) the SCS, and (3) the CACS, each of which can provide long-term RHR. The SCS and the CACS are separate and independent. The MLCS is used for normal power operation and for all RHR modes, including a DBDA. The SCS may be used for RHR following reactor shutdown for all but a very limited number of low probability accident initiators (i.e., DBDA). The CACS may be used for RHR following any normal or emergency shutdown of the reactor. Table 4-7 summarizes the design requirements for these three RHR systems. Figure 4-15 shows the general cooling system arrangement.

The principal safety function of the three cooling systems is to transfer heat from the reactor core to the ultimate heat sink. Under full-load operation, the heat transferred from the reactor core is ~1088 MW(t). The MLCS is capable of extended operation ranging from 100% down to 25% for electric power production. The MLCS is also used for RHR following reactor trip. The SCS and CACS are designed to provide core cooling only following reactor trip.

The secondary function of the three cooling systems is to maintain the structural components inside the PCRV at temperatures at which they can safely and efficiently perform their intended functions under all normal and accident situations throughout the reactor design life.

The PCRV liner, penetrations and closures, parts of the steam generators, the main and auxiliary helium circulators, and the reactor mechanisms housings make up the pressure-retaining boundary of the reactor coolant system. The function of the boundary is to contain the reactor coolant during all normal and abnormal temperature and pressure conditions and to confine any radioactive material and limit its accidental release to acceptable values.

TABLE 4-7
RHR SYSTEM CAPABILITIES

	MLCS	SCS	CACS
Number of loops	3	3	3
Seismic class	Not applicable	I	I
Power source	On site	Off site	On site
	Off site	Off site	Off site
		1E	IE
Safety grade/ seismic class	No/no	Yes/1	Yes/1
System capability			
Pressurized	1 out of 3 loops	1 out of 3 loops	2 out of 3 loops
Depressurization accidents	2 out of 3 loops	--	2 out of 3 loops
Design basis for normal refueling	2 out of 3 loops	--	2 out of 3 loops
Refueling under abnormal condition(a)	1 out of 3 loops	2 out of 3 loops	1 out of 3 loops
Natural convection pressurized	Not applicable	Secondary side only	2 out of 3 loops
Repressurized natural convection at refueling	Not applicable	Secondary side only	2 out of 3 loops

(a)Not to exceed the PC-5 core temperature limits shown in Section 5.1.2.1

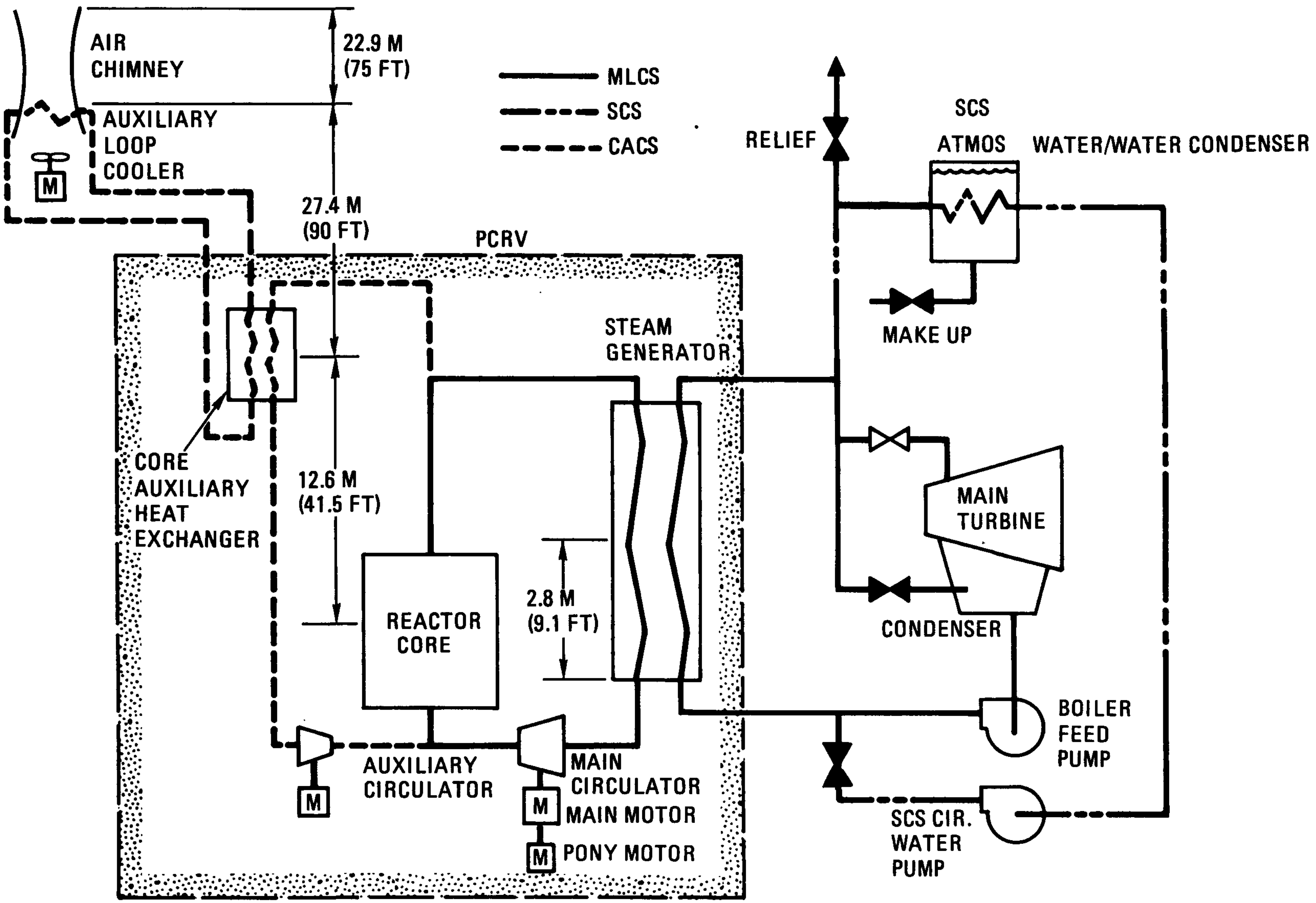

Fig. 4-15. GCFR heat removal systems

Since the cooling loops and their components are all located in PCRV cavities, the PCRV cavity liners and the penetration liners and closures provide the ultimate reactor coolant boundary. The PCRV provides structural support to withstand the coolant pressure. Internal ducts that provide the helium flow paths between the reactor, steam generators, and circulators are also located in the PCRV for both the MLCS and CACS.

The reactor coolant system, in conjunction with its control and protective provisions, will be designed to accommodate the system pressures and temperatures under all modes of plant operation, including anticipated transients and postulated accidents.

The safety design bases for GCFR RHR are derived to adequately assure that acceptable fuel cladding and pressure boundary temperatures are maintained for all credible events which lead to reactor shutdown. The key elements for the GCFR safety design bases are the following:

1. Two redundant safety systems, the CACS and the SCS, shall be provided for long-term RHR.

2. The CACS and the SCS shall be seismic category I.

3. The SCS and the CACS shall be independent from each other.

4. The reliability goal for the RHR function shall be such that the probability of loss of design core cooling geometry shall be beyond the design basis.

5. Natural circulation RHR capability shall be adopted with appropriate experimental verification.

The CACS system is designed to meet all the requirements of the Institute of Electrical and Electronic Engineers (IEEE) Standards 279 and 603 (Refs. 4-1 and 4-2); Regulatory Guide 10CFR50, Appendix A (Ref. 4-3); and Nuclear Regulatory Guides related to safety systems. The SCS meets the same requirements as the CACS, except for some failures which are caused by design basis events that remove a main cooling loop from service. These design basis events can be accommodated by the CACS while meeting the single failure criteria of industry standards and NRC requirements.

The overall purpose of the RHR systems is to provide adequate cooling of the shutdown reactor core for all plant conditions created by normal operation and accident events within the plant design basis. Control functions are incorporated into each system to enable it to fulfill this purpose.

The following specific control functions are required for each of the three forced convection RHR systems:

1. Perform the necessary sequencing functions to establish the desired process flow paths and to activate and bring the proper system equipment on line.

2. Provide sufficient control to maintain all process transients within the acceptable operating limits that have been established for the involved equipment with due consideration for the operating environment (under accident failure and normal conditions) and the frequency and probability of occurrence of the particular event.

3. Establish and maintain the desired process operating setpoints and ranges where these factors are necessary for the proper process system functioning.

4. Perform an orderly shutdown of the system or system parts to a standby or off state.

5. Provide system testing functions to verify safety system operability.

No control functions are required for natural convection RHR operation.

RHR system control has three aspects: (1) a sequence of events (or open-loop control) required to bring the system on line, (2) the process control (or closed-loop control) necessary for the system to perform its intended function, and (3) the capability to compensate for some performance degradation and system failures.

4.5.1. MLCS

4.5.1.1. Design Bases. Main loops are sized for power generation (Section 4.3.1.1). However, a number of main loop characteristics and special requirements enhance shutdown cooling. In general, the main loop equipment is not safety class or seismic category I; exceptions are the following:

1. Main loops have the thermal-hydraulic capability to provide adequate core cooling for all design basis events, including the [194 cm^2 (30 $in.^2$)] DBDA. The main loops are designed to tolerate the containment environment for a DBDA. The main loops are the preferred cooling system for all events that do not incapacitate the main loops. In general, one of three loops is capable of RHR (100% capability); however, for low-frequency events, such as DBDA, two of three loops are required (50% capability).

2. The main circulator motor can isolate itself from nonsafety power supplies and freely coast down. This coastdown capability is a safety-class function and provides RHR in the critical seconds following reactor and/or main loop trip, because of the large rotating inertia of the motor. Heat can be transferred safely to the steam generator for this period even with no feedwater flow.

3. When the CACS is operating, the main loops shall shut down and isolate themselves. A helium-side MLIV is designed to close by gravity and reverse flow from other operating circulators and is

provided with a fail-safe actuator for closing only. (This is a safety-related feature.) If the valve still fails to close, the layout and elevations of main loop equipment shall cause the loop to self-isolate on the helium side, so as not to interfere with CACS natural circulation. (Self-isolation means that the hydrostatic head created by the column of relatively cool helium opposes and balances the core differential pressure established by natural convection CACS operation, such that no backflow occurs through the main loops.)

4. Other features (described in Section 4.3.1) which enhance power generation reliability also enhance shutdown cooling reliability.

4.5.1.2. System Description. Section 4.3.1.2 describes the MLCS during normal operation. RHR-mode operation is very similar, except as follows.

During startup and shutdown operation, three main turbine bypass steam systems are used, one for each steam generator. Each bypass has a desuperheater, flash tank, steam bypass lines to the main condenser, and associated controls. The bypass lines, in turn, supply low pressure steam through the flash tank to the auxiliary steam headers. Steam can also be supplied to the auxiliary headers by the auxiliary boilers. This system provides steam to drive the boiler feedwater pumps following reactor trip and steam to heat steam lines and feedwater heaters.

4.5.1.3. RHR Initiation and Termination System.

MLCS RHR Intiation. The RHR function of the MLCS is automatically initiated by the primary (Section 4.4.1) or secondary (Section 4.4.2) reactor trip systems. The RHR function will normally be performed by the MLCS, the system that is on line when plant shutdown is initiated. The MLCS is designed to have full shutdown and RHR capability.

The MLCS RHR can also be initiated manually by the plant operator for a normal plant shutdown.

Section 4.5.1.4 discusses operation and control for the above conditions.

MLCS Loop Shutdown System. Figure 4-16 shows a typical (one of three) MLCS loop shutdown system. The following section discusses the initiating conditions and instrumentation and logic system that initiate shutdown of an individual (or all) malfunctioning MLCS loop(s).

MLCS Loop Shutdown Initiating Conditions. The MLCS loop shutdown system automatically initiates shutdown of an individual malfunctioning loop, based upon parameter measurements within the loop or all loops, upon receiving a signal indicating that either the SCS (Section 4.5.2.3) or the CACS (Section 4.5.3.3) have started. The following conditions cause an individual MLCS loop to be shut down:

1. Circulator power off.

2. High circulator speed (>115%).

3. Circulator bearing pressure low (later).

4. Loop helium outlet temperature high [>338°C (>640°F)].

5. Loop steam temperature high [>566°C (>1050°F)].

6. Loop steam pressure low [<8.27 MPa (<1200 psia)] with the reactor at power (>20%).

7. Loop feedwater flow low (<20%, 20 s delay) with the reactor at power (>10%).

MLCS Loop Shutdown System Logic. (See Fig. 4-16). The MLCS loop shutdown system consists of three redundant instrument channels and two redundant logic systems to initiate shutdown of a malfunctioning main loop. The system is configured so that a single failure of a component or module will

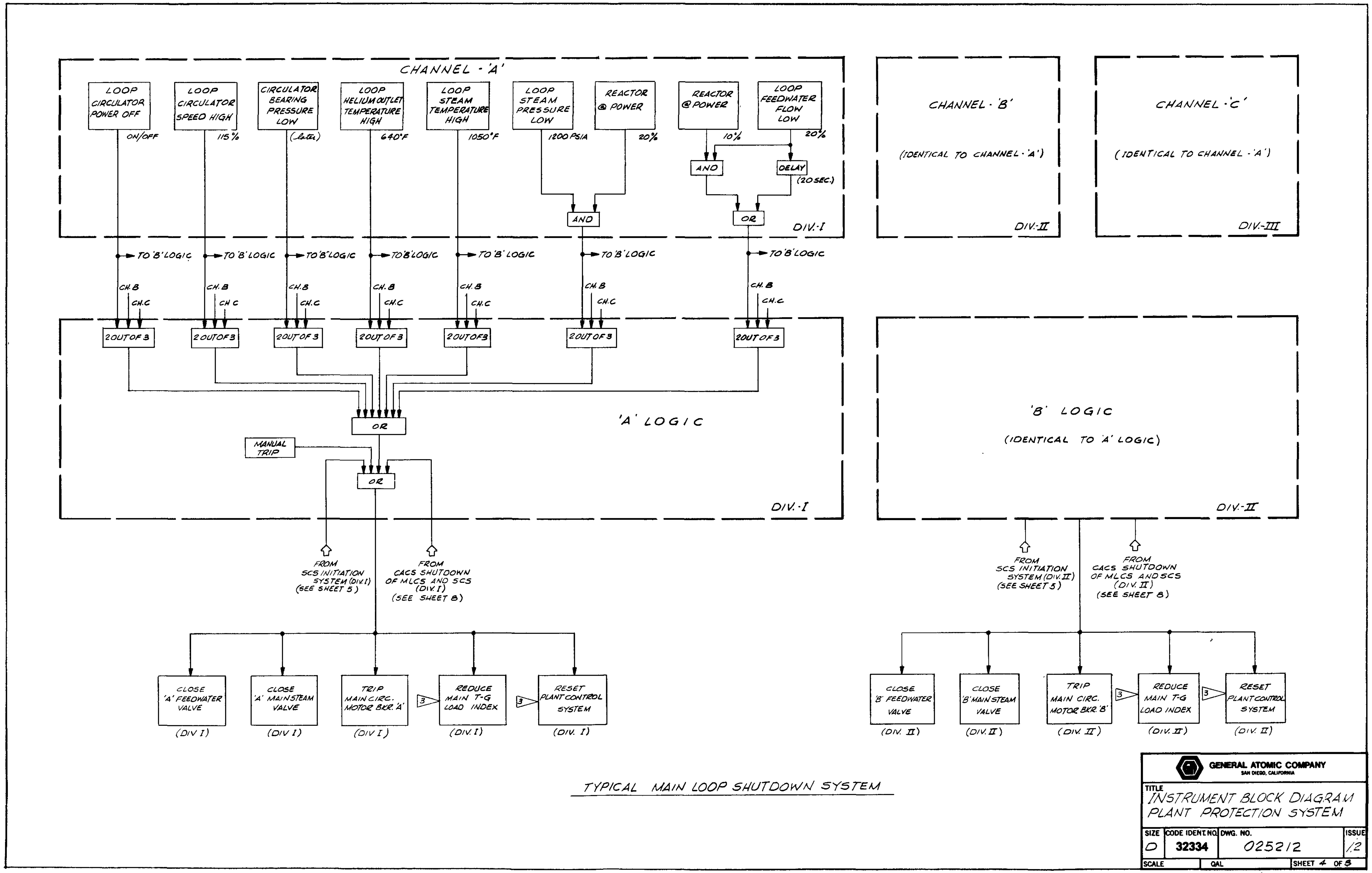

Fig. 4-16. Typical main loop shutdown system

only affect the operation of one MLCS loop. Additionally, no single random failure in the shutdown system will prevent the shutdown of the malfunctioning loop.

Three redundant instrument channels monitor the operation of each MLCS loop. Each instrument channel contains sensors, process instrumentation, bistables, and logic to initiate trip signals to seven independent, two-out-of-three logic trip detectors in each of the two logic systems.

The 14 trip detectors are arranged to form an "A" and a "B" logic output. The "A" logic will initiate shutdown of the malfunctioning loop with "A" logic and "A" actuators. The "B" logic will initiate shutdown of the malfunctioning loop with "B" logic and "B" actuators. Additionally, each logic system contains a manual trip input and inputs from the SCS initiation system (Section 4.5.2.3) and from the CACS shutdown of MLCS and SCS system (Section 4.5.3.3). The latter two inputs are transmitted to all three MLCS loop shutdown systems and initiate shutdown of all MLCS loops.

Any time two of the three instrument channels sense a malfunctioning loop, the instrument channels will transmit trip signals to a two-out-of-three detector in each logic system. The two-out-of-three detectors will trip and, in turn, transmit signals to shut down the malfunctioning loop.

In addition to shutting down an individual loop, the MLCS loop shutdown system sends signals to the plant control system to reduce the main turbine/generator load index and to reset the plant control system for continued operation at reduced load, if appropriate.

4.5.1.4. Operation and Control. Following a reactor trip, the MLCS will automatically be reconfigured to perform the RHR function. The specific events are automatically sequenced in a set pattern, and processes are controlled to predetermined values. Plant operator intervention in the initial sequence is limited because of the relatively fast time frame in which these events must occur.

For a normal station shutdown, the plant operator will sequence the events required to establish the RHR operating mode. A major difference between the automatic sequencing and the normal operator-controlled shutdown is the timing of the events. In the normal shutdown, reactor and station power are gradually reduced and all process circuit reconfigurations are considerably stretched out in time. Since the operator is initiating the events, he has some flexibility to depart from the set automatic sequence. For example, the operator can have auxiliary steam ready before reactor shutdown is complete and begin phasing it in much earlier than is done under the automatic sequence.

Table 4-8 gives the MLCS automatic transition sequence from the normal station power production configuration to the RHR configuration. This sequence is initiated by a reactor trip (Sections 4.4.1 and 4.4.2). The transition occurs in two stages, distinguished only by the source of low pressure auxiliary steam. In the first stage, the main steam generators supply the auxiliary steam headers with low pressure steam through the desuperheaters and flash tanks, as shown schematically in Fig. 4-17. The second stage begins when the auxiliary boilers start to supply steam to these headers. By this time, the steam generators are flooded out and are being used as helium-to-water heat exchangers. Figure 4-18 illustrates the process flows for this latter stage. This is the long-term RHR configuration for the MLCS.

If a reactor trip occurs from a plant condition other than normal power production (e.g., reactor power below 20%, turbine off-line), the transition sequence will start at the appropriate point, based on the existing plant configuration and corresponding process conditions.

An additional supplementary aspect of this sequence is the action necessary to shut down the main loop plant components not used in this RHR mode of operation (e.g., turbine generator) or to bring them to a desired standby state. Although this action is not necessary to effect the transfer from normal operation of the MLCS to RHR operation, conditions for these

TABLE 4-8
MLCS RHR CHRONOLOGICAL TRANSITION SEQUENCE

Stage	Component Identification	Event	Remarks
0		Reactor trips	Initiation sequence
First	See Fig. 4-17[a]	Rapid runback; main turbine generator trips; valve V1 closes	Runback rate determined by need to conserve steam capacity of steam generators
		Main steam bypass circuit activates; valves V2, V3, V4, and V6 open as required	V2 controls pressure at the steam generator exit, V3 controls the desuperheater exit temperature, V4 controls the auxiliary steam header pressure by-passing excess steam to the condenser, and V6 controls the flash tank water level.
		Ramp feedwater flow to 25% and helium circulator speed to 30%	Ramp rates adjusted to provide acceptable process transients. Minimum 15% feedwater flow is required to ensure steam generator boiling stability during floodout.
		Auxiliary boilers start	
Second		Auxiliary steam source switches to auxiliary boiler; valve V5 opens (valve V4 has closed)	When adequate auxiliary boiler steam pressure is available and flash tank steam production decreases, auxiliary steam header pressure is maintained by V5 at a slightly lower pressure than that maintained by V4.

TABLE 4-8 (Continued

Stage	Component Identification	Event	Remarks
Third		Long-term RHR conditions are established; process control set-points are set to long-term RHR values.	

[(a)]For simplicity, Figs. 4-17 and 4-18 show only one of the three MLCS loops. The loops that are not shown are schematically the same in all respects.

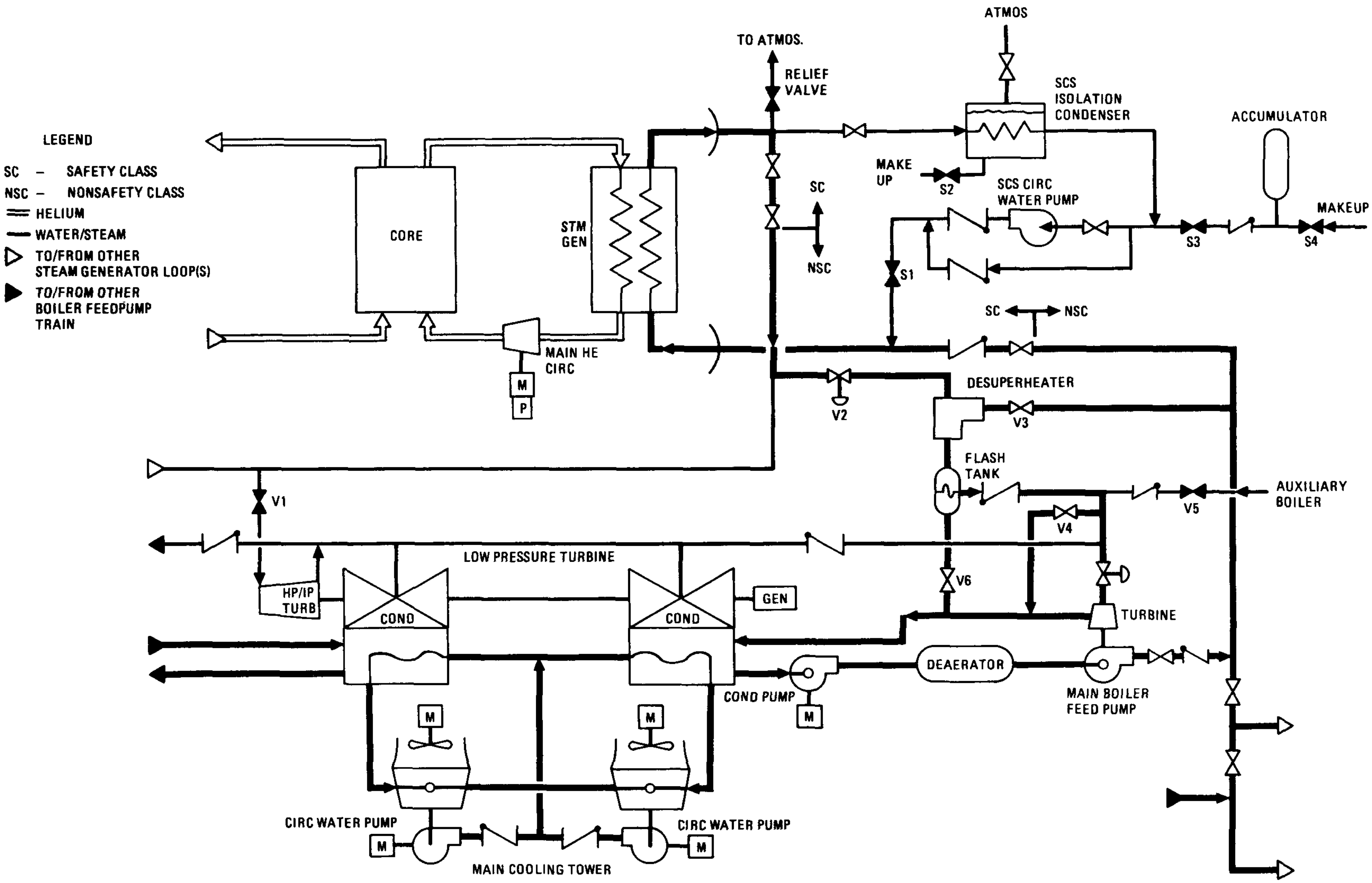

Fig. 4-17. GCFR main and shutdown cooling systems, first-stage MLCS RHR, process flow

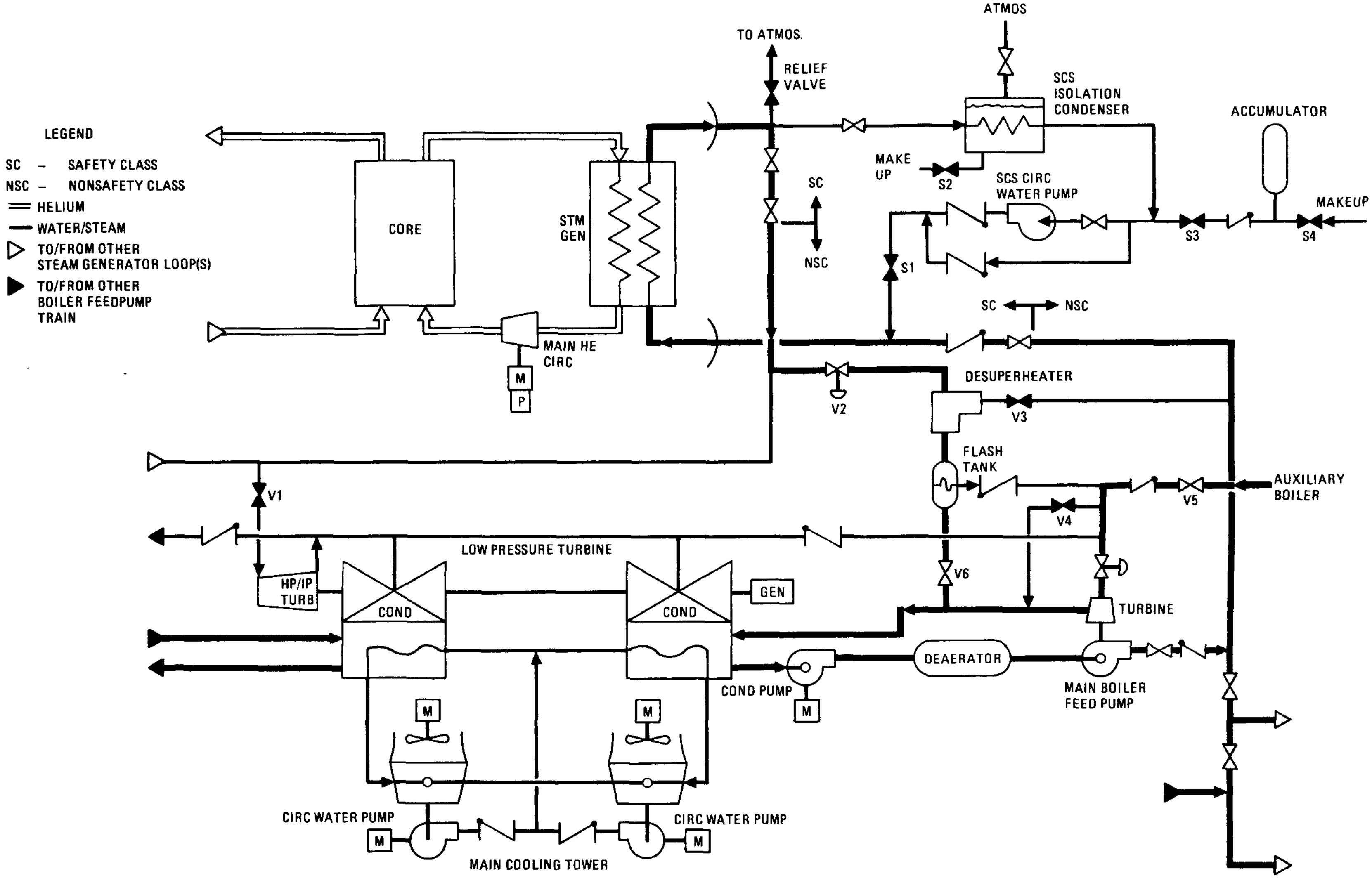

Fig. 4-18. GCFR main and shutdown cooling systems, long-term MLCS RHR, process flow

components should be maintained within acceptable limits, and the plant availability should be maximized.

The sequencing steps can be thought of as open-loop control. Closed-loop (automatic feedback) control is also needed to maintain the desired process conditions and to provide acceptable process transient conditions. The three major automatic control loops used during MLCS RHR operation are (1) steam generator exit pressure regulation, (2) feedwater flow regulation, and (3) helium flow regulation.

Bypass valve V2 (see Figs. 4-17 and 4-18) regulates the steam generator exit pressure, providing adequate boiling behavior in the steam generators during floodout and minimizing transient conditions imposed on the steam generator. This control loop remains active until steam generator floodout has been determined to be complete. The bypass valve is then set to a fixed position for long-term water flow.

The same flow control loop utilized for the 25% to 100% load range regulates the steam generator feedwater flow. This control loop provides an appropriate feedwater flow rate to the steam generator. Following reactor trip, the feedwater setpoint is ramped to 25% of full flow. The flow is held there unless the operator changes this setpoint.

Helium flow regulation uses the circulator speed control subsystem from the normal on-load control system. This loop provides adequate helium flow to maintain core internal temperatures within acceptable limits. Circulator speed is preprogrammed to attain the helium flow requirement for adequate cooling. Speed, then, is the direct control variable. The speed setpoint is an inverse function of PCRV static pressure, with the value set at 30% of the full reactor power speed level at an inlet plenum pressure of 9.5 MPa (1400 psia). The motor speed increases in response to a PCRV depressurization to compensate for the decreasing mass flow resulting from decreasing coolant density.

The main circulator motor design requirements are based on plant conditions prevailing during normal station operations. Therefore, their ability to provide adequate gas flow during and after a depressurization event is a function of core pressure drop, containment back pressure, rate of depressurization, and motor torque/speed characteristics. However, in all cases, the inverse relationship between the demanded motor speed and the PCRV pressure produces the desired motor response.

For controlling both the feedwater flow and helium flow, normal on-load plant control loops are used in conjunction with the control effectors and instrumentation. However, because of the significant changes in system operating conditions, the on-load controller configuration must be switched to an RHR configuration.

Corrective action can be taken to compensate for some MLCS failures and performance degradation. The feedback control loops will automatically provide some corrective action within the limits of their capability. The operator can also make certain compensating adjustments.

Under normal conditions, the MLCS RHR mode provides substantial overcooling capability. If a failure eliminates one loop, each of the other loops has sufficient capability to adequately perform the RHR function. For the loss of a single loop, the following actions are taken:

1. The remaining loops continue to operate at the same loop feedwater flow and circulator speed.

2. The failed loop is shut down in an orderly fashion.

The loop feedwater flow need not be adjusted, since the shutdown feedwater flow rate (25%) is based on boiling stability and is adequate to remove the core decay heat without the loop(s) that failed.

If the system performance degrades to the extent that PPS limits are exceeded (Section 4.5.1.3), then the SCS or CACS will be brought on-line to perform the core cooling function, and MLCS action will be terminated.

4.5.2. SCS

4.5.2.1. Design Bases. The SCS is an ESF which backs up the MLCS for cooling the shutdown reactor and for removing decay heat produced by the core. As such, the SCS shall meet the criteria of Section 3.3 (e.g., redundancy, independence, single failure, reliability, diversity, etc.). The SCS shall be used when the MLCS is not available or when MLCS RHR use is undesirable. The SCS shall rovide long-term RHR for those more frequent events which would otherwise limit total plant core coolng capability. The only accidents which the SCS is not designed for are (1) feedwater or steam line breaks inside the containment, (2) depressurization events in which the primary helium pressure decreases below 2.07 MPa (300 psia), (3) core disruptive accidents, (4) anticipated transients without scram (ATWS), and (5) design basis natural phenomena other than the safe shutdown earthquake (SSE).

For pressurized events [helium pressure greater than 2.07 MPa (300 psia)], each loop (one of three) shall be capable of adequate RHR. (These events are PC-2.)

Following normal shutdown and depressurization (such as for maintenance or refueling), two loops shall be capable of adequate RHR. (These events are PC-3.)

After some time has elapsed and core decay heat has fallen sufficiently, one loop shall be adequate for RHR. (The SCS can be used for refueling after reactor shutdown if necessary, but it would be an unusual occurrence and not a design basis event. (The MLCS or the CACS is normally used for refueling.)

These requirements can change, depending on reliability analysis results.

The SCS duty cycles shall be determined later. The SCS components have the following limiting sizing conditions:

1. Main circulator (compressor). Sized by power generation.

2. Pony motor. Speed and power determined to be adequate for pressurized RHR and for maintaining the auxiliary loop isolation valves in a closed position.

3. Steam generators. Sized by power generation.

4. Shutdown cooling water system (SCWS). Sized by pressurized cooling events.

4.5.2.2. System Description. The SCS backs up the MLCS for cooling the shutdown reactor and removing decay heat produced by the core. The SCS is one of two independent and diverse RHR safety-class systems; the other system is the CACS (described in Sections 4.5.3 and 4.5.4).

The SCS will be used when the MLCS is unavailable or when MLCS RHR use is undesirable. The SCS provides long-term RHR for all but a limited group of extremely low probability accident initiators. The SCS must provide adequate cooling to prevent the temperatures of the fuel, the cladding, and the reactor internals from exceeding prescribed limits, so that safe cooldown of the reactor is ensured after any credible combination of simultaneous system failures.

The SCS consists of three independent and separate loops. Figure 4-19 shows a typical loop. Each loop has two heat transfer circuits: (1) the primary coolant circuit and (2) the secondary water circuit with heat ultimately rejected to the atmosphere. The SCS shares the main circulator,

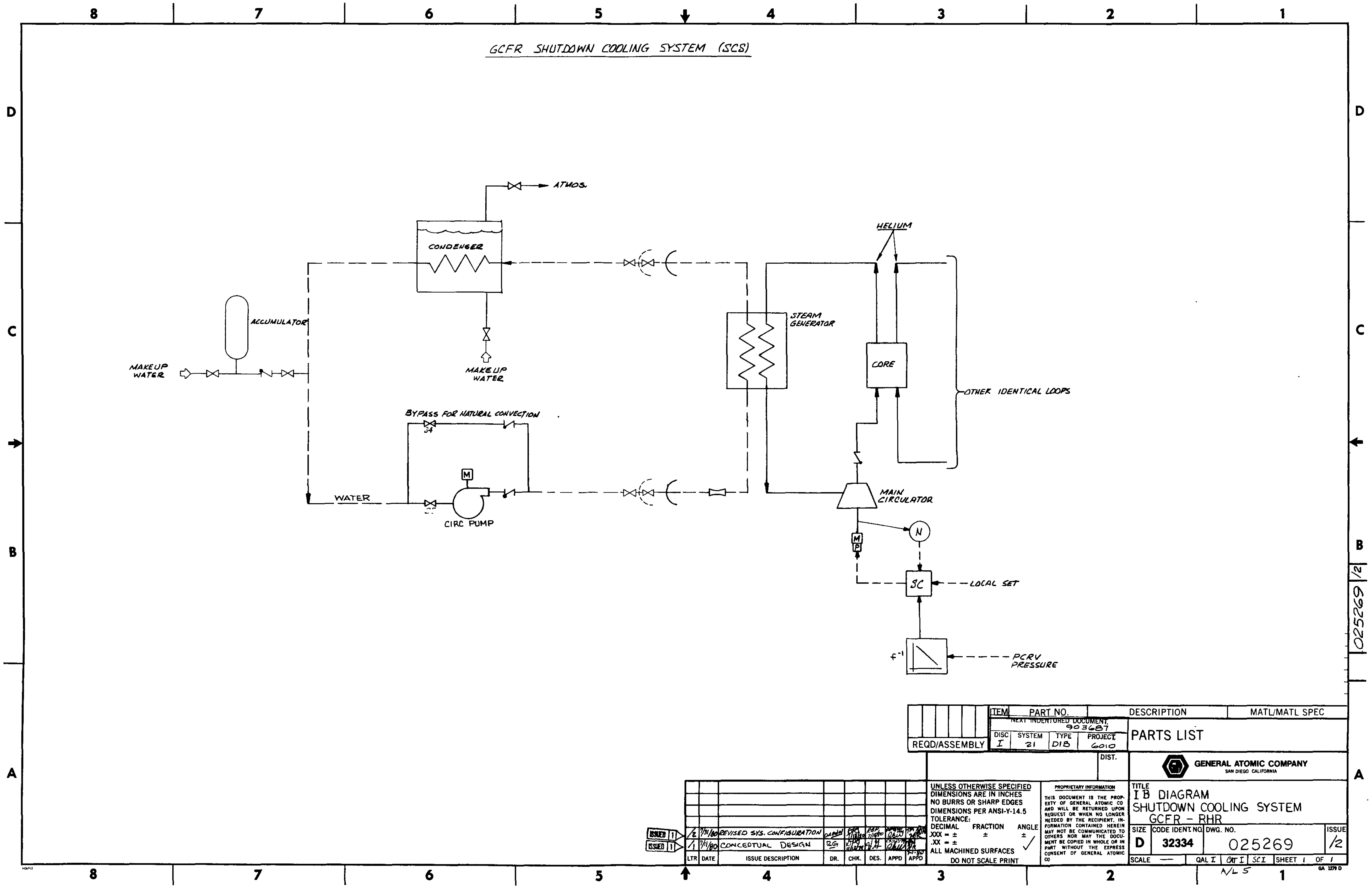

Fig. 4-19. Typical SCS configuration

drive shaft, and steam generator with the MLCS. The SCS includes a pony drive and protected power source to drive the main circulator.

The secondary water circuit consists of a water-cooled condenser, a motor-driven circulating water pump, and interconnecting piping with associated valves. In each loop, water or steam circulates between the main steam generator and the condenser. The water circuit operates with forced convection, but it has potential for natural convection.

Helium circulated by the pony-driven main helium circulators removes heat from the core and transfers it to the water/steam in the steam generators. The water/steam circulates to the condenser, where the heat is transferred to the cooling water in the condenser drum. The heat input to the drum results in gradual vaporization of the stored water in the drum. The drum is maintained at atmospheric pressure, with the generated steam exhausting to the atmosphere. The condensed water in the tubes is circulated back to the steam generators as feedwater.

The isolation condenser is a multitubed water-to-water heat exchanger with a large water drum providing a heat sink. Sufficient water storage is maintained in the drum to permit ~30 min of passive cooldown after a reactor trip. For extended SCS cooldowns, make-up water is supplied to the drum to maintain the water level.

The SCS will be used for RHR under a number of accident conditions, including LOSP, loss of on-site power, and loss of feedwater. Section 4.5.2.4 presents the sequence of events for transferring to the SCS.

4.5.2.3. RHR Initiation and Termination.

SCS Initiation System. Figure 4-20 shows the SCS initiation system. The following section discusses the initiating conditions and the instrumentation and logic system that initiate startup of SCS loops and shutdown of the MLCS loops. The SCS backs up the MLCS for core cooling.

SCS Initiating Conditions. The initiation system automatically initiates SCS loop startup and MLCS shutdown any time that the MLCS loops are not operating in a mode to adequately cool the reactor core. The following conditions cause the SCS loops to automatically start up:

1. Low normal feedwater flow (<20%) to all three main loop steam generators (20 s delay) with the reactor pressurized.

2. Low circulator speed (<28%) in all three loops with the reactor pressurized.

3. Logic signal from the primary reactor trip system indicating high primary coolant moisture.

SCS Initiation System Logic. (See Fig. 4-20.) The SCS initiation system consists of three redundant instrument channels, three trip detectors to initiate the startup of the three SCS loops, and six trip detectors to initiate the shutdown of all three main loops. The system is arranged in a coincidence logic configuration to ensure that a single failure of a component or module will affect only the operation of one SCS or MLCS loop. With sufficient independence in the configuration shown, no single random failure in the initiation system will prevent the startup of at least two SCS loops and the shutdown of all MLCS loops.

The three redundant instrument channels monitor the operation of the three MLCS loops to determine if they are operating in a mode that will cool the core. Each instrument channel contains sensors, process instrumentation bistables, and logic to initiate trip signals to the nine independent two-out-of-three logic trip detectors.

The six trip detectors, which shut down all the MLCS loops, are arranged in a configuration to form an "A" and a "B" logic output. The "A" logic contains three of the two-out-of-three trip detectors, and the "B" logic contains the other three two-out-of-three trip detectors. The "A" logic will initiate shutdown of all MLCS loops with "A" logic and actuators.

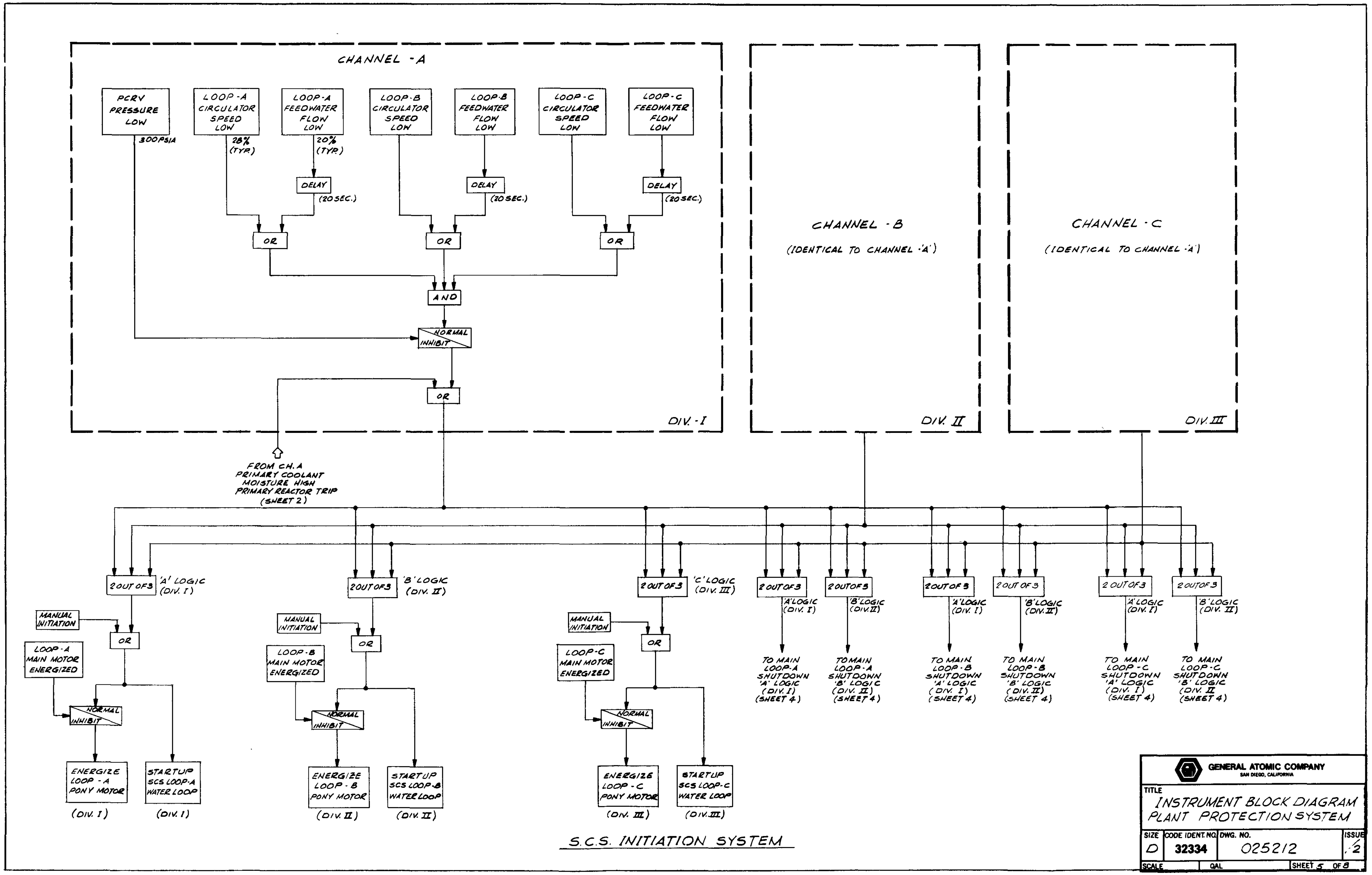

Fig. 4-20. SCS initiation system

The "B" logic will similarly initiate shutdown of all MLCS loops with the "B" logic and actuators.

Any time two of the three instrument channels sense that all three MLCS loops are not operating in a mode to adequately cool the reactor core, the instrument channels will transmit trip signals to all trip detectors. The two-out-of-three detectors will trip and, in turn, transmit signals to start up all three SCS loops and shut down all three MLCS loops.

SCS Loop Shutdown System. Figure 4-21 shows a typical SCS loop shutdown system. The following section discusses the initiating conditions and instrumentation and logic system that initiate shutdown of an individual (or all) malfunctioning SCS loops(s).

SCS Loop Shutdown Initiating Conditions. The SCS loop shutdown system automatically initiates shutdown of an individual malfunctioning SCS loop, based upon parameter measurements within the loop, or shuts down all three loops upon receiving a signal indicating that the CACS has started. The following conditions cause an individual SCS loop to be shut down:

1. Circulator bearing pressure low (late) and the pony motor energized (10 s delay).

2. SCS loop water pressure low [<3.45 MPa (<500 psia)] and the water loop operating (20 s delay).

3. SCS loop water flow low (<20%) and the water loop operating (20 s delay).

SCS Loop Shutdown System Logic. (See Fig. 4-21.) The SCS loop shutdown system consists of three redundant instrument channels and two redundant logic systems to initiate shutdown of a malfunctioning SCS loop. The system is configured so that a single failure of a component or module will only affect the operation of one SCS loop. Additionally, no single random failure in the shutdown system will prevent the shutdown of the malfunctioning SCS loop.

Three redundant instrument channels monitor the operation of each SCS loop to determine if it is operating properly. Each instrument channel contains sensors, process instrumentation, bistables, and logic to initiate trip signals to six independent, two-out-of-three logic trip detectors.

The six trip detectors are arranged in a configuration to form an "A" and a "B" logic output. The "A" logic will initiate shutdown of the malfunctioning SCS loop with "A" logic and actuators. The "B" logic will initiate shutdown of the malfunctioning SCS loop with "B" logic and actuators. Because the SCS is a closed safety class system, only the "A" logic closes the loop isolation valves. Additionally, each logic system contains a manual trip input and inputs from the CACS shutdown of the MLCS and SCS (Section 4.5.3.3). This latter input is transmitted to all three SCS loop shutdown systems and initiates shutdown of all SCS loops.

Any time two of the three instrument channels sense a malfunctioning loop, the instrument channels will transmit trip signals to the two-out-of three detectors in each logic system. The logic detectors will trip and, in turn, transmit signals to shut down the malfunctioning loop.

4.5.2.4 Operation and Control. The transition sequence from MLCS operation to SCS cooling is fully automatic and is initiated by the PPS or the plant operator. Initiation of SCS cooling also shuts down all MLCS loops (Section 4.5.2.3). THe MLCS water/steam paths are isolated by closing valves V7, V8, V9, and V10 (see Fig. 4-22). Valve S1 is opened, and the SCS motor-driven circulating water pump is actuated and provides 25% feedwater flow. Should the SCS circulating water pump not start, potential for natural circulation exists on the water side through the pump and through a bypass around the pump.

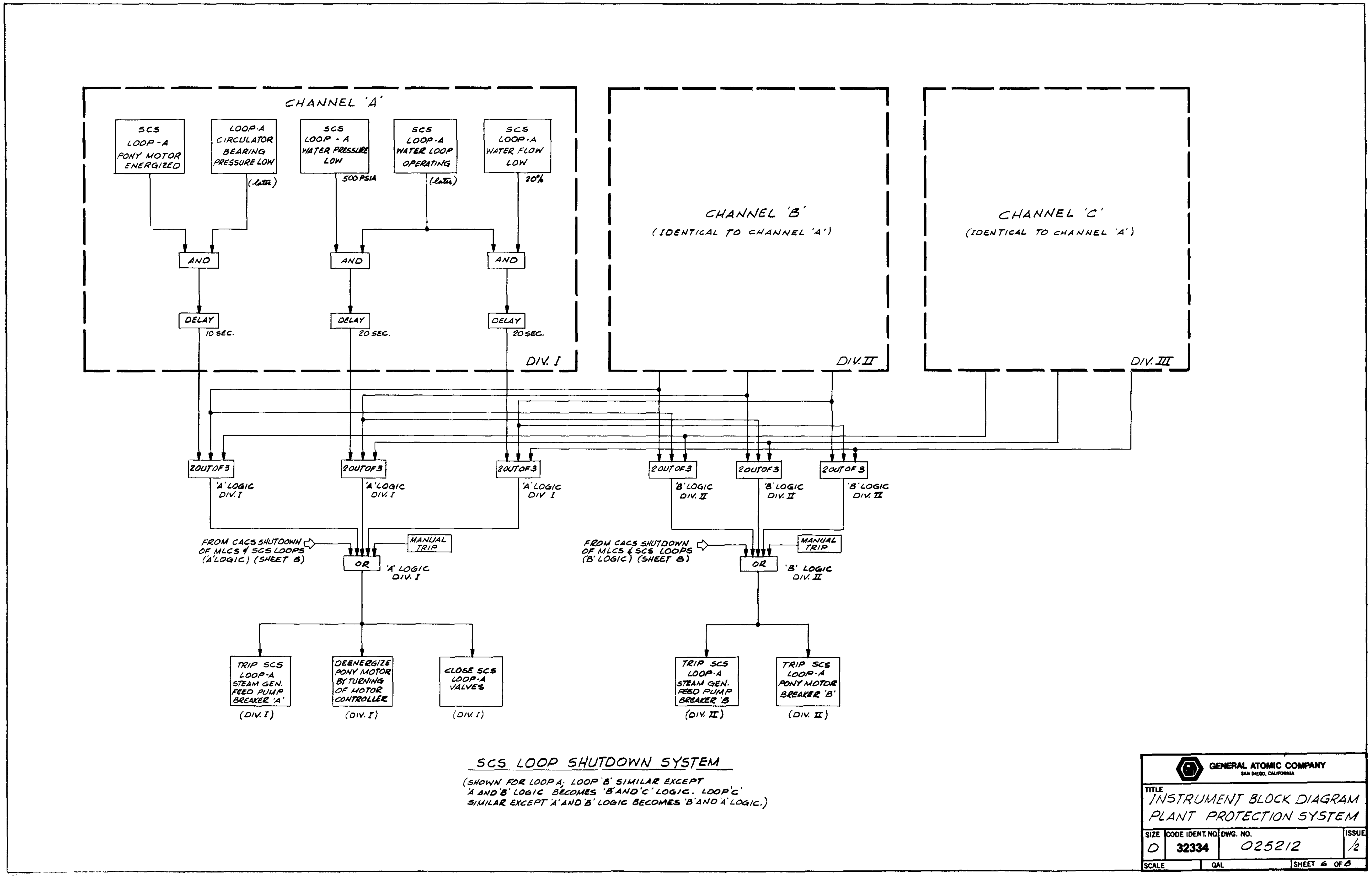

Fig. 4-21. SCS loop shutdown system

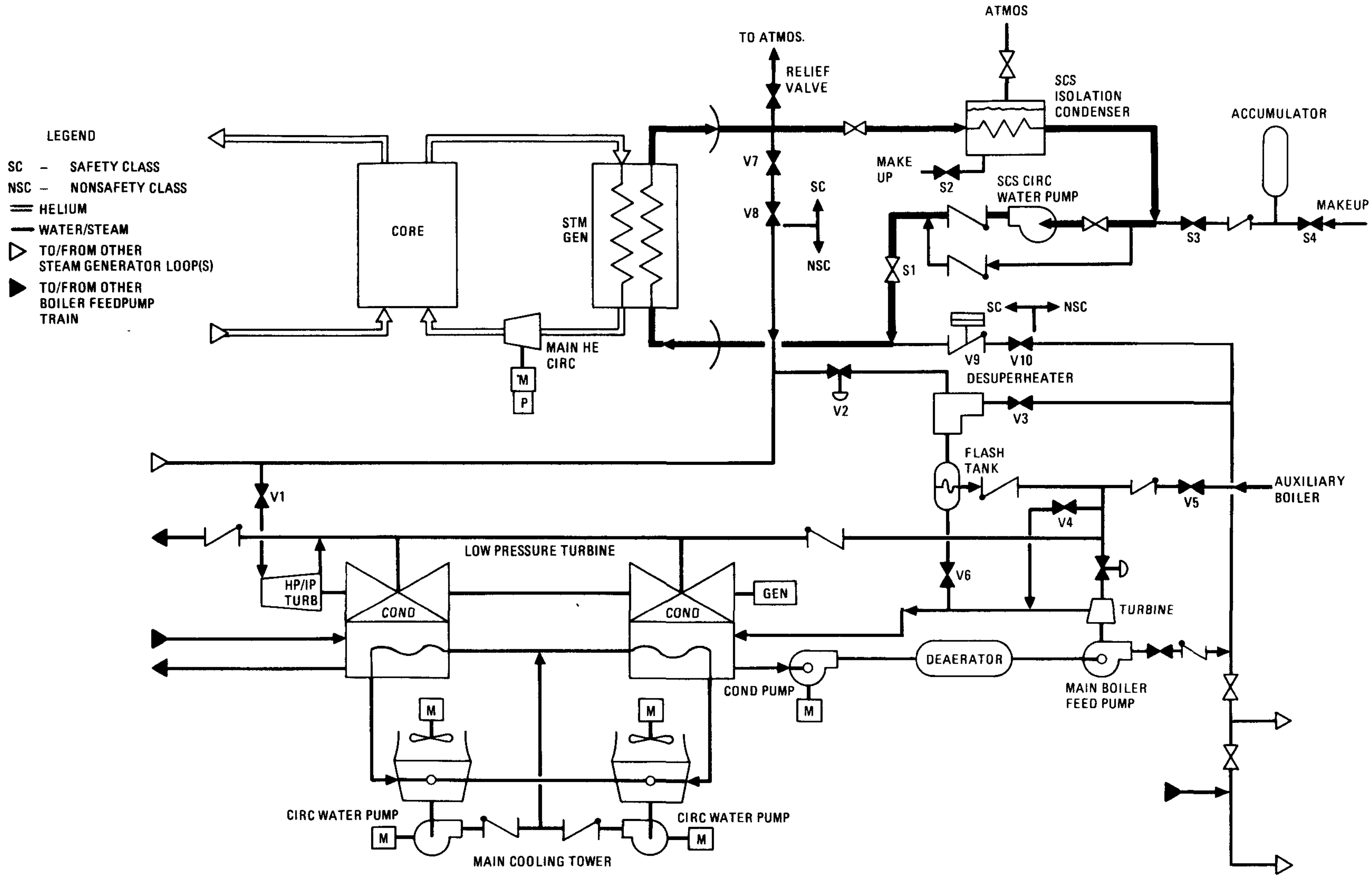

Fig. 4-22. GCFR main and shutdown cooling systems, SCS RHR, process flow

Helium flow is regulated in essentially the same manner as the MLCS RHR function. The pony motor has a speed controller which synchronizes motor rotation to the coasting circulator shaft, phases in the circulator load, and regulates the motor speed about a setpoint (30%). The control system also responds to a PCRV depressurization event in the same way as the MLCS controller. However, the SCS pony motor is limited because of a maximum of 50% speed and, therefore, cannot handle depressurization accident events.

Essentially, the same corrective actions are taken to compensate for SCS failures and performance degradations as described for the MLCS RHR function. This includes feedback control loop action and the capability to continue adequate core cooling with only one loop in the event of the complete functional failure of one or two of the other loops.* However, the initial activation and cooling action is completely automatic, and operator intervention is limited to the long-term RHR. If the system performance degrades to the extent that PPS limits are exceeded, the CACS will be brought on line, and SCS operation will be terminated. After an appropriate set of conditions have been satisfied, indicating that adequate core cooling is available and that the core decay heat generation is sufficiently low, the operator can terminate SCS operation and initiate either the CACS or the MLCS.

The SCS isolation condenser is sized to permit ~30 min of passive cooldown after a reactor trip. For long-term SCS RHR, the following actions are necessary: (1) open valve S2, providing makeup water to the isolation condenser, (2) open valve S3 to maintain the SCS loop pressure, and (3) open valve S4 to maintain the accumulator water level.

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*Each SCS loop is capable of providing 100% of the required core cooling capability pressurized. This means that overcooling will occur when all loops are operating and that system operating adjustments are not necessary when one or two loops become inoperative.

4.5.3. CACS

4.5.3.1. Design Bases. The CACS is an independent, safety-class ESF, which cools the shutdown reactor and removes the core decay heat. As such, it meets the criteria of Section 3.3 (e.g., redundancy, independence, single failure, reliability, diversity, etc.). The system is used (1) when the MLCS and the SCS loops are unavailable, (2) when the MLCS is unavailable during a DBDA, or (3) when MLCS or SCS RHR use are undesirable. The CACS shall provide safe cooldown of the reactor during pressurized conditions with two loops operational. The CACS shall provide safe cooldown during PCRV depressurization accidents with leak areas up to and including 200-cm^2 (30-$in.^2$) with any two of the three loops operational.

The CACS components are sized by an iteration procedure, since tradeoffs can be made (for example, between flow rate and heat exchanger surface area). However, the components are primarily sized as follows:

1. The CACS circulator/motor is sized primarily by the DBDA with air ingress.

2. The water pump and air loop cooler (ALC) fans are primarily sized by pressurized cooling events with forced circulation.

3. The ALC/core auxiliary heat exchanger (CAHE) surface area and elevation differences are sized primarily by pressurized natural circulation cooling events.

The system is additionally designed to be used during refueling, as described in Section 4.3.2.1, with two loops operating. Section 4.5.4.1 discusses design bases for the CACS pressurized and repressurized in the natural circulation mode.

The CACS helium-side loop has an auxiliary loop isolation valve (ALIV), which is a check valve opening by gravity or forward flow from the auxiliary circulator. This valve is also provided with a fail-safe actuator for opening only.

The CACS duty cycles shall be determined later.

4.5.3.2. System Description. The CACS provides an independent means of cooling the shutdown reactor and removing the core decay heat. The CACS must provide adequate cooling and prevent the temperatures of the fuel, the cladding, and the reactor internals from exceeding prescribed limits so that safe cooldown of the reactor is ensured after any credible combination of simultaneous system failures.

The CACS is a safety-class system designed to perform the engineered safety function of providing adequate core cooling, operating as a pressurized or depressurized forced circulation system. The CACS is also designed to provide adequate core cooling for a pressurized PCRV by natural circulation flows in both the gas and water loops. This latter operating mode significantly enhances the pressurized cooling reliability and diversity (see Section 4.5.4).

In addition to the engineered safety function, the core auxiliary cooling loops provide RHR during refueling.

The CACS must also provide safe reactor cooldown with a mixture of gases in the reactor coolant system, such as could result from accidents involving steam or air ingress.

The CACS consists of three separate and independent cooling loops. Each loop contains a CAHE, an auxiliary circulator, and a loop isolation valve, located in a PCRV cavity, and a pressurized cooling water loop, located primarily outside the PCRV.

Each auxiliary circulator is electrically driven by an independently powered motor. The circulator motors are contained within the reactor coolant envelope and are protected from excess temperature by thermal insulation and redundant water cooling systems. The ALIV is mounted in the inlet duct of the auxiliary circulator. It is a split butterfly valve actuated to open either by gravity and/or by the gas flow differential pressure. When the auxiliary circulator is shut down and the main circulators or other auxiliary circulators are running to provide forced circulation, the ALIV is closed by the reverse pressure difference. When forced circulation fails, the valve opens by gravity to allow forward natural circulation flow. In addition, a fail-safe actuator is available to open the valve if gravity actuation fails. The CAHE employs helically coiled tubes with helium passing over the tubes in a single-pass, cross-counterflow.

The core auxiliary cooling water system (CACWS) is a pressurized water loop thermally coupling the CACS with its ultimate heat sink, the atmosphere. The system will be used whenever the auxiliary loops are employed for core cooling.

The CACWS works in conjunction with the CACS for engineered safety and refueling functions. Component sizing is sufficient to reject any anticipated heat loads on a CACS loop without exceeding the water saturation temperature.

The CACWS consists of three independent pressurized water loops (see Fig. 4-23 for piping layout). Each loop circulates water between a CAHE and its respective ALC. In addition to the interconnecting piping and heat exchangers, each loop has a pressurizer and a motor-driven circulating water pump with associated valves.

The CACWS is designed to operate with either forced circulation or natural circulation. For forced circulation operation, an electrically driven centrifugal pump circulates the cooling water. Electrical power is available from multiple sources, including the essential power bus. Section 4.5.4.2 discusses natural circulation.

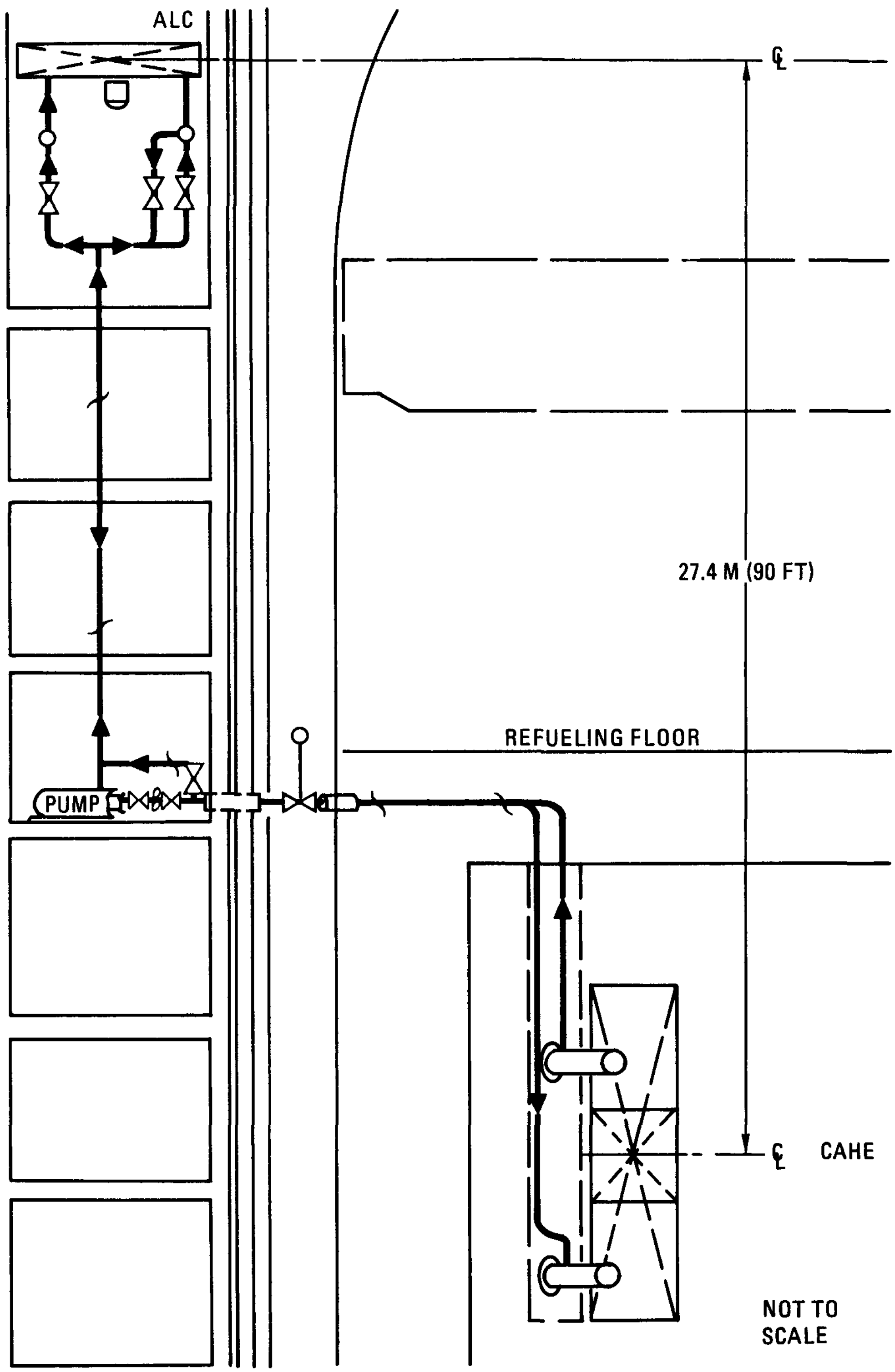

Fig. 4-23. CACWS piping: (a) elevation view (sheet 1 of 2)

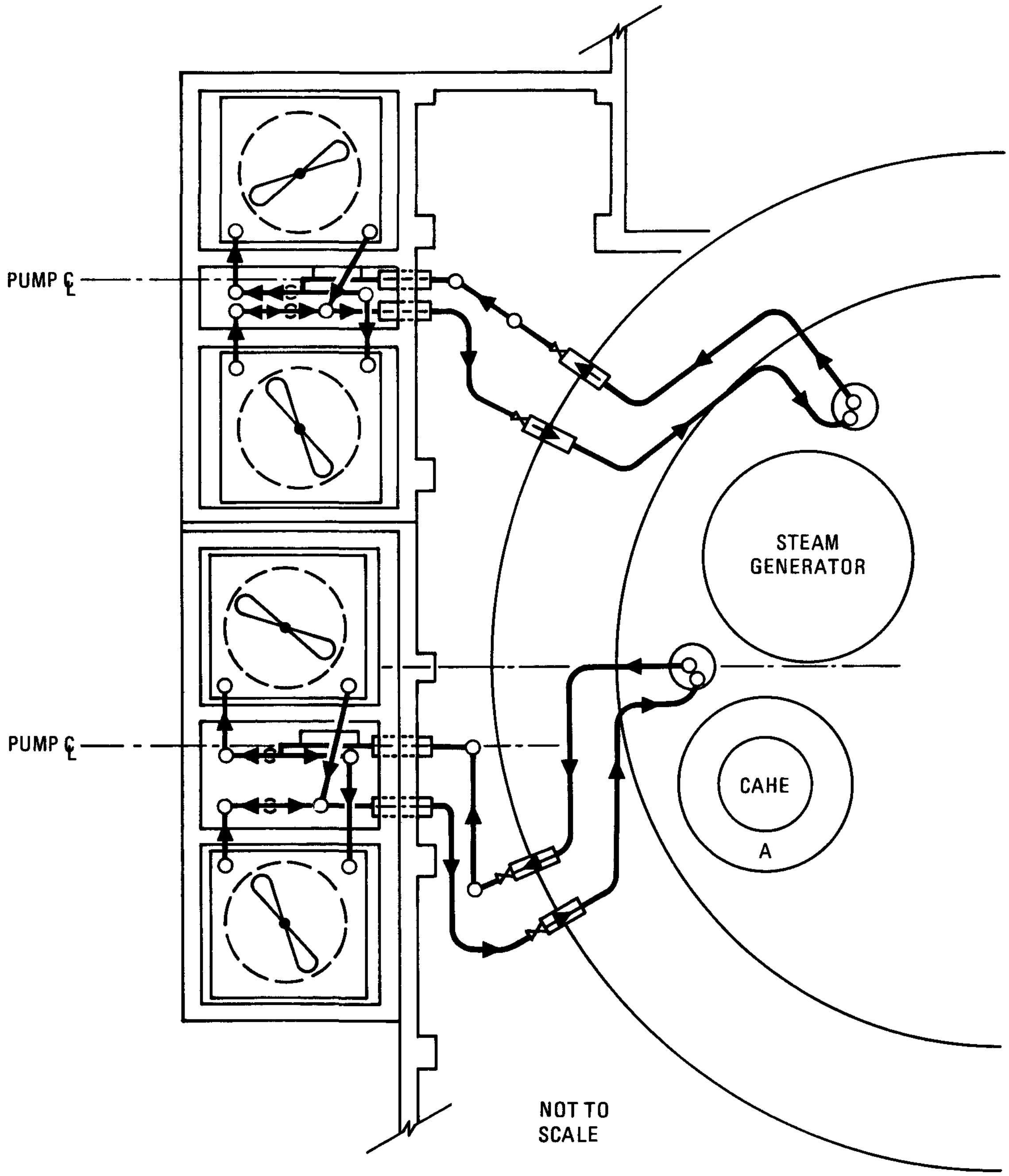

Fig. 4-23. CACWS piping: (b) plan view (sheet 2 of 2)

Natural circulation of cooling water between the CAHE and ALC during normal plant operation maintains the system in a standby condition and prevents CAHE overheating. Heating under these conditions is small and is due to a designed helium back-leakage from the core inlet plenum through the closed ALIV.

The ALC is a finned, multipass, water-air heat exchanger. Three such units are located in the upper corners of the confinement building. Natural draft for each heat exchanger is provided by a 23-m (75-ft) chimney. Table 4-9 gives detailed design data for the CACS system.

Water pressure of 9.31 MPa (1350 psia) is maintained and boiling is prevented by a pressurizer with a nitrogen accumulator. At this pressure, boiling would occur at ~306°C (582°F).

The auxiliary loops are designed and instrumented to facilitate periodic testing of operability while the plant is generating electrical power. Since the heat exchanger and related heat removal system will be in continuous service with circulation of cooling water, no special testing for operability is necessary. The ALIVs are periodically cycled to ensure operability. The circulating water pump will also be periodically operated against a closed discharge valve to verify availability.

4.5.3.3. RHR Initiation System and MLCS/SCS Shutdown System.

CACS Initiation System. Figure 4-24 shows the CACS initiation system. The CACS is brought into action if the SCS and MLCS are not performing adequately. In addition, the CACS is used for safe shutdown cooling in certain low probability events (e.g., DBDA) not included in the SCS design bases. The following sections discuss the system initiating conditions, then the system logic.

TABLE 4-9
CACWS CONFIGURATION

Parameters	CAHE	Hot Leg Pipe	ALC	Cold Leg Pipe
Tube o.d. [m (ft)]	0.0287 (0.094)	0.405 (1.33)	0.0265 (0.087)	0.405 (1.33)
Tube i.d. [m (ft)]	0.0235 (0.077)	0.363 (1.19)	0.0216 (0.071)	0.363 (1.19)
Tube length [m (ft)]	14.4 (47.3)	80.2 (263)	31.9 (104.7)	86.6 (284)
Vertical leg length [m (ft)]	--	25.5 (83.5)	--	25.5 (83.5)
Number of tubes	200	1	494	1
Frontal area [m^2 (ft^2)]	4.18 (45.0)	--	112.7 (1213)	--
Number of passes	--	--	12	--
Heat transfer area [m^2 (ft^2)]	259.2 (2790) (Installed)	--	18,720 (201,500)	--
Type	Helical	Insulated	Finned	Insulated
Tube side ΔP loss coefficient, K_{tot}	16.5	7.8	4.1	11.8
Air cooling tower height [m (ft)]	--	--	22.9 (75)	--
air flow loss coefficient	--	--	21.2	--
Conservative ΔP uncertainty factor				
H_2O	1.2	1.2	1.2	1.2
Air	1.2	1.2	1.2	1.2

For the CACS to be fully effective, the main cooling loops must be shut down. Therefore, once the CACS is on line, a feedback signal shuts down both the SCS and MLCS. This is described later in this section.

CACS Initiating Conditions. The initiation system automatically starts up the CACS loops and shuts down the SCS and MLCS loops any time the MLCS and SCS loops are not operating in a mode to adequately cool the reactor core or when certain other events occur that require CACS operation.

The following conditions initiate CACS:

1. High primary coolant pressure [>11.34 MPa (>1645 psia)].

2. Low primary coolant pressure [<2.07 MPa (<300 psia)] and main helium circulator speed of less than 80% in two out of three loops.

3. Low primary coolant pressure [<2.07 MPa (<300 psia)] and low feedwater flow (<20%) in two out of three main steam generators [can be bypassed if steam generator helium inlet temperature is <427°C (<800°F)]. Flow delayed (80 s) to allow SCS to start if it is not on line.

4. All three main circulator speeds less than 20%.

5. Low feedwater flow (<20%) to all three main steam generators [can be bypassed if core helium outlet temperature is <427°C (<800°F)]. Flow delayed (80 s) to allow SCS to act if it is not on line.

CACS Initiation System Logic. (See Fig. 4-24.) The CACS startup initiation consists of three redundant instrument channels and three redundant two-out-of-three trip detectors to initiate the startup of the three CACS loops. The startup system is arranged in a coincidence logic configuration

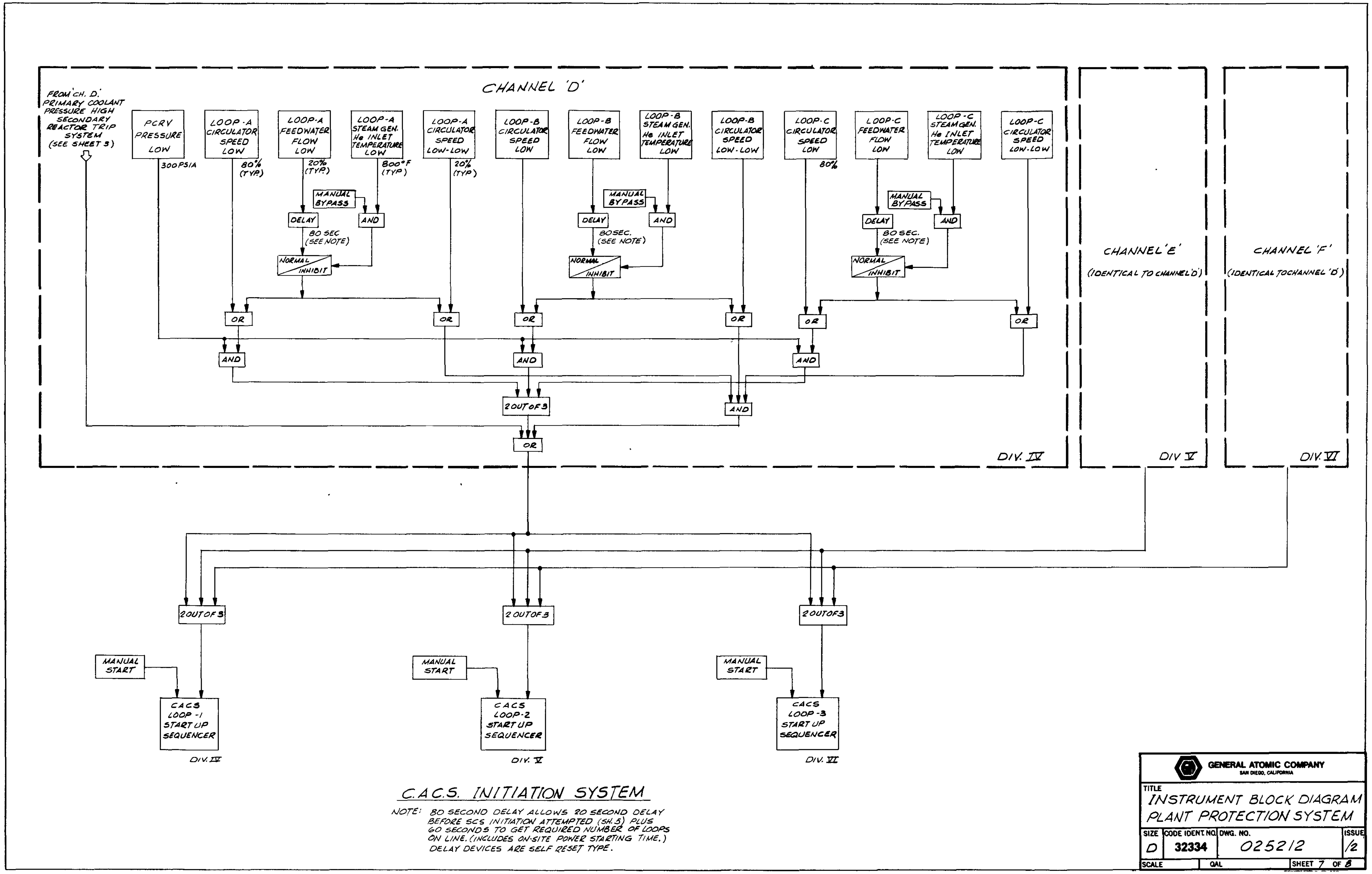

Fig. 4-24. CACS initiation system

so that a single failure of a component or module will not accidentally initiate CACS loop startup or prevent the initiation of a startup signal to all three CACS loops. The overall CACS system is designed so that no single random failure will prevent the startup of at least two of the three CACS loops.

Each of the three redundant instrument channels contains sensors, process instrumentation, bistables, and logic to monitor the process parameters and to transmit output trip signals from each monitored parameter.

The output from the OR gate of each instrument channel is fed to three redundant two-out-of-three trip detectors to energize the CACS loop sequencers.

The instrument channels have a bypass to allow the reactor core to be continually cooled by the SCS or MLCS. The low feedwater flow through the steam generators function may be bypassed with the PCRV pressure high or low. To insert the bypass, the operator must manually operate a switch in each instrument channel, and the core helium outlet temperature must be <427°C (<800°F). If the temperature rises above 427°C (800°F), the bypass is automatically removed and the trip function reinstated.

CACS Shutdown of MLCS and SCS Loops. The CACS performance is dependent upon shutdown of the main cooling loops. To assure that main loop cooling is not prematurely terminated, this action is not initiated until the CACS cooling is initiated and available. Figure 4-25 shows the CACS initiation of SCS and MLCS shutdown. Subsequent sections describe the initiating conditions and logic involved.

CACS Shutdown of Main Loops, Initiating Conditions. The PPS automatically shuts down the SCS and MLCS loops when one* or more CACS loop is brought on line. The initiating conditions for this are as follows:

1. Adequate normal CACS loop cooling water flow [>45.5 kg/s (>100 lb/s)].

2. CACS circulator speed >5% (100% speed is 3600 rpm).

CACS Shutdown of Main Loops, Logic. (See Fig. 4-25).) The shutdown system contains three redundant instrument channels, three redundant logic systems to initiate the automatic shutdown of all SCS loops, and two redundant logic systems to initiate the automatic shutdown of all MLCS loops when one or more CACS loop is brought on line. The system is designed with a coincidence logic configuration so that a single random failure of a component or module will only affect a single shutdown train of one loop, except a power supply failure, which may affect one logic system of each loop.

Three redundant instrument channels monitor the three CACS loops and send trip signals to redundant logic systems any time one or more CACS loop is started up and placed on line. Each instrument channel contains sensors, process instrumentation, and bistables to generate trip signals to 12 two-out-of-three logic trip detectors in the redundant logic systems.

The 12 two-out-of-three trip detectors are arranged into a logic configuration to form three redundant logic systems called "A", "B", and "C" for the SCS and two systems called "A" and "B" for the MLCS. The logic systems contain the two-out-of-three trip detectors that generate trip signals to shut down the SCS (Section 4.5.2.3) and MLCS (Section 4.5.1.3) loops. The logic is arranged to prevent multiple loop shutdown due to a single active failure of a component or module.

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*Tentative

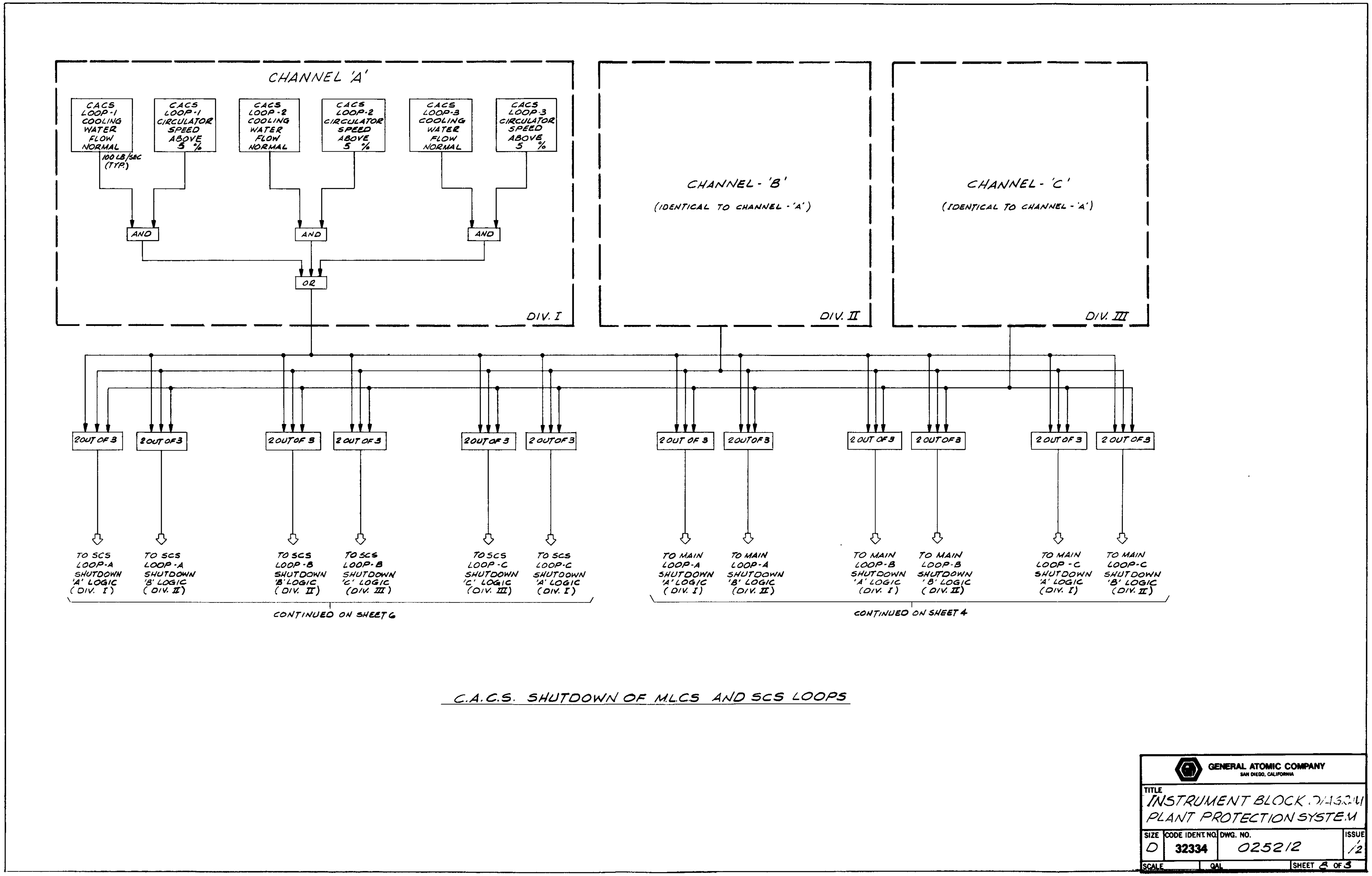

Fig. 4-25. CACS shutdown of MLCS and SCS loops

Any time two of the three instrument channels sense that one or more CACS loops have been started up and brought on line, the instrument channels will transmit trip signals to all 12 two-out-of-three trip detectors. The two-out-of-three trip detectors will trip and, in turn, transmit signals to shut down all three SCS and MLCS loops.

4.5.3.4. Operation and Control. The safety-class CACS is completely independent from the other two RHR systems. When not performing its core cooling function, it is maintained in a standby mode, as shown in Fig. 4-26.* Natural convection provides sufficient flow to maintain water chemistry and the desired water temperature.

Table 4-10 gives the activation sequence for the CACS core cooling function. This sequence is fully automatic and is initiated by the PPS or the plant operator. Operator intervention after system initiation is limited to the long-term cooling. Figure 4-27 shows the RHR process flows.

Transition from MLCS or SCS core cooling to CACS cooling will occur without any specific control action when the pressure rise generated by the CACS circulators exceeds the decreasing pressure rise from the coasting-down main circulators. The auxiliary and main helium shutoff valves will then open and close, respectively, and the helium flow will be channeled through the auxiliary loops by either the auxiliary circulators or by natural circulation.

The CACS is designed to require minimum control action for all design basis events. Based on this design, the only control needed is water temperature regulation during standby operation and helium flow regulation.

During the RHR mode, all fan motors are activated, and they remain that way until operator intervention is permitted during the long-term RHR. The

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*For simplicity, the figures in this section (Figs. 4-26 and 4-27) show only one of the three CACS loops. The CACS loops that are not shown are schematically the same in all respects.

LEGEND

T = WATER TEMPERATURE, COLD LEG
TC = TEMPERATURE CONTROLLER
M = ELECTRIC MOTOR
N = ROTATIVE SPEED
SC = SPEED CONTROLLER

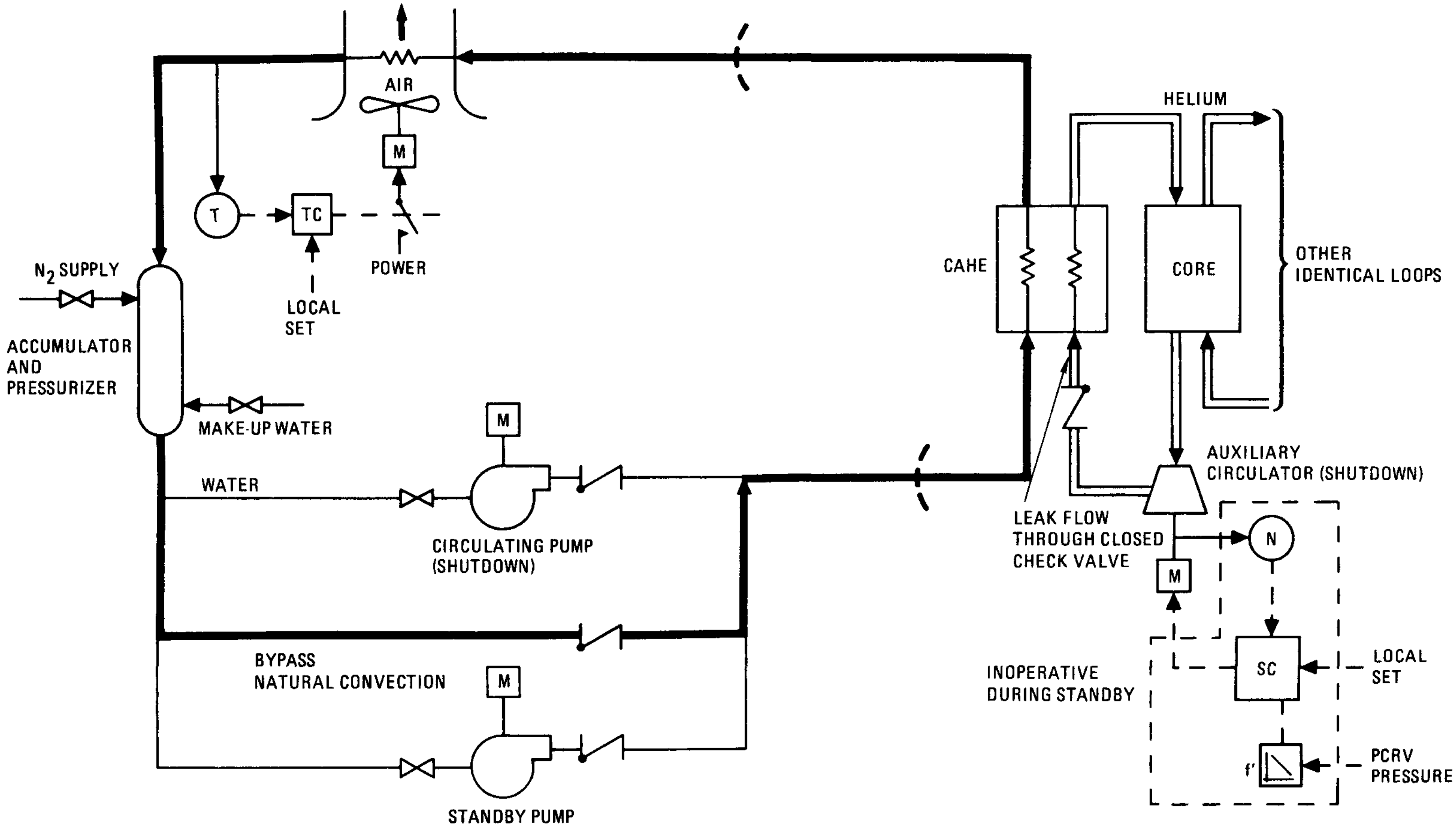

Fig. 4-26. GCFR CACS, standby mode

TABLE 4-10
CACS ACTIVATION CHRONOLOGICAL SEQUENCE

Stage	Event[a]	Remarks
0	Activation signal appears	Signal induced by PPS or operator
1	CACS circulator motors activate	
2	Full CACS water flow initiates; circulating pump energizes	Motors undergo acceleration to full speed
3	Full air flow through air/water heat exchanger initiates; all air fan motors activate	
4	After confirmation of stage 1 action, the MLCS and SCS are shut down	Main circulator shafts are allowed to coast down without any braking

[a] See Figs. 4-26 and 4-27 for component identification.

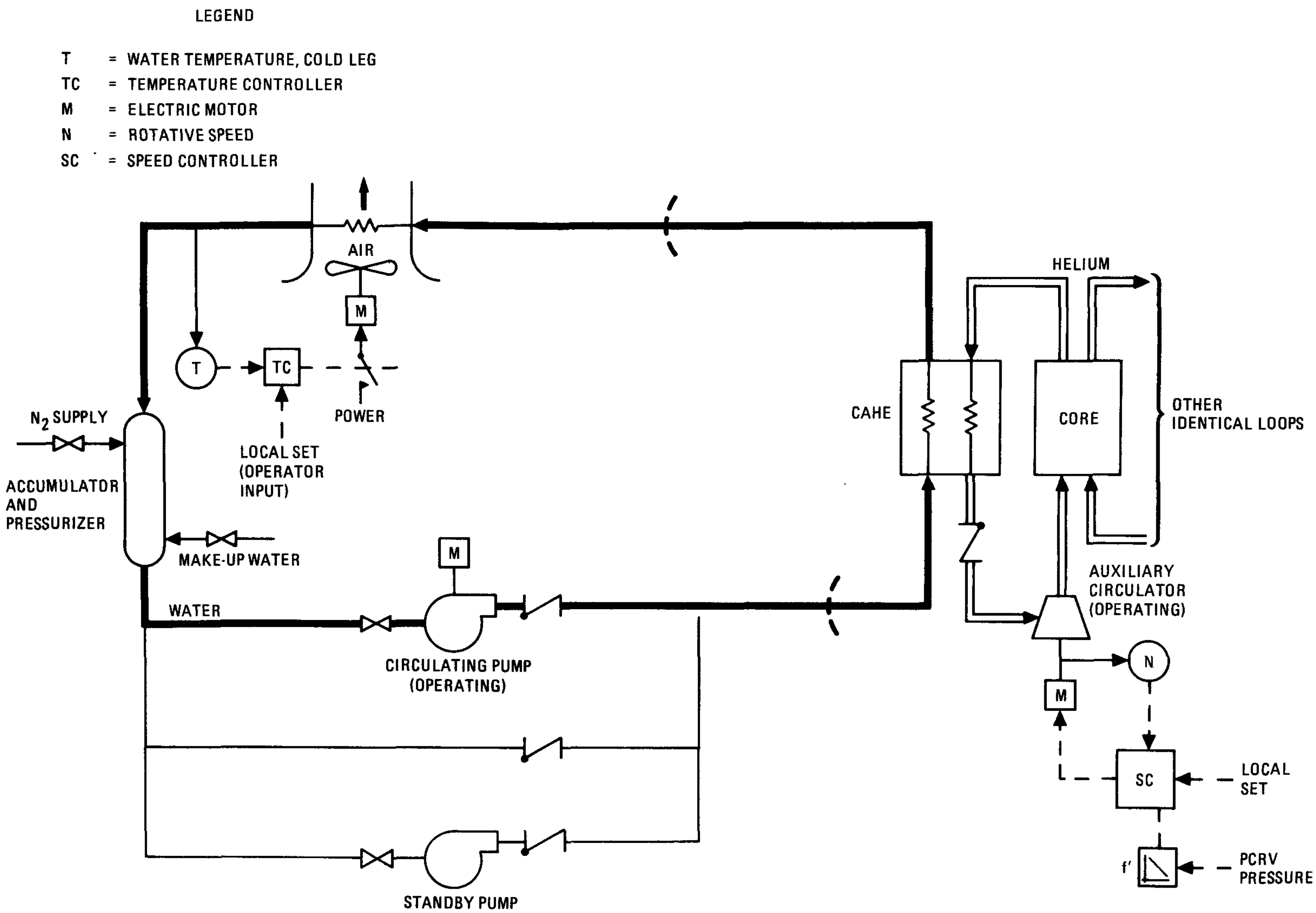

Fig. 4-27. GCFR CACS, RHR mode

operator can then selectively turn off fan motors to adjust the water temperature and indirectly adjust the helium temperature.

Helium flow control is used to compensate for the changing helium conditions during a PCRV depressurization. Helium flow control as a function of PCRV pressure is accomplished in essentially the same manner as described for the MLCS and SCS RHR functions. A motor speed setpoint equivalent to ~10% of CACS motor design speed is used when the PCRV is fully pressurized. The motor speed is increased in response to a depressurization event. By design, the CACS circulator motors have torque/speed characteristics that are adequate for the full range of depressurization events from fully pressurized to the DBDA. This flow control is necessary because of large density changes that occur on the helium side.

Limited corrective actions are available to compensate for CACS failures and performance degradations. It is intended to be a very basic and straightforward operating system with a minimum of closed-loop control required. The system failure tolerance is obtained because of its single-failure-proof design and its independence from other plant RHR systems. The system can adequately cool the core under all design basis conditions with the complete loss of one of the three loops. Also, 2 h after reactor trip from full power, one loop is sufficient to provide adequate cooling. The CACS is overdesigned to the extent that no system adjustments are required to continue adequate cooling following failure of one loop.

The initial activation and cooling action is completely automatic, and operator intervention is limited to long-term RHR. After an appropriate set of conditions has been satisfied, indicating that adequate core cooling is available and that the core decay heat generation is sufficiently low, the operator can terminate CACS operation and initiate either the SCS or MLCS. Only the plant operator can terminate CACS operation.

4.5.4. Natural Circulation CACS

4.5.4.1. Design Bases. The CACS is designed to provide passive short- and long-term natural circulation RHR to the ultimate heat sink. This ESF is fully integrated into the NSS and BOP plant design. Natural circulation provides a diverse backup to the three forced circulation RHR systems described previously. However, the GCFR is designed so that deterministic licensing requirements are fully satisfied by the three forced circulation systems. The CACS provides adequate natural circulation cooling under pressurized conditions using two loops.

For situations where the primary system has been depressurized for refueling or maintenance, the plant design allows for repressurization within 15 min to a level where natural circulation cooling will adequately cool the core using one CACS loop. To enhance the licensability of the natural circulation CACS, the following general points shall be reflected in the plant design:

1. Provide adequate elevation difference between the core and the CAHE, within the PCRV. Select the relative elevations of the core, CAHE, and steam generator and the duct configurations to provide a self-isolation feature to limit bypass flow in the event that the MLIVs fail to close.

2. Provide simple flow paths to minimize concern for secondary flows or thermal traps.

3. Assure that under normal plant operation the temperature profiles established in the CACS (both NSS and BOP portions) are favorable to the startup of natural circulation.

4.5.4.2. System Description. Section 4.5.3.2 describes the CACS and its forced circulation RHR capability.

The CACS design also incorporates natural circulation capabilities on the helium, water, and air sides as a backup to normal forced circulation capabilities. Core decay heat is transported by the primary coolant helium to the pressurized water-cooled CAHE, which is elevated 12.6 m (41.5 ft) above the core. Heated water from the CAHE reaches the ALC, located 27.4 m (90 ft) above the CAHE, by natural circulation in the pressurized water loop. The heat from the ALC is ultimately rejected to the atmosphere by natural draft of air through a 22.9-m (75-ft) chimney. For a total loss of forced circulation capability, the natural circulation capability of the CACS provides cooling for an indefinite period if the primary coolant is pressurized.

Refueling and maintenance are conducted under depressurized conditions using slightly subatmospheric helium. For higher probability depressurized events, such as refueling, the GCFR is designed so that, upon loss of one of two forced circulation systems, the PCRV can be rapidly repressurized to a pressure level where adequate primary coolant natural circulation is available. Figure 4-28 schematically shows the repressurization system. Helium is supplied from the normal helium storage system, which consists of 120 tanks of helium, each ~12 m (~40 ft) long and 0.6 m (2 ft) in diameter. For PCRV repressurization, tanks are connected through a header to a common 15-cm (6-in.) schedule 160 line which branches to enter the PCRV at two locations in the lower plenum. Valves in the line provide for storage system and containment building isolation. When the PCRV is depressurized, ~18,144 kg (40,000 lb) of helium are in the storage tanks at a pressure of 10.34 MPa (1500 psia). If emergency repressurization is required during refueling, the reactor isolation valve is closed and the normally closed valves in the repressurized line are opened. The helium in the storage system discharges through the 15-cm (6-in.) line into the PCRV lower plenum. When PCRV pressure attains ~1.72 MPa (250 psia), the helium inventory is sufficient to maintain adequate core cooling via natural convection only.

The reactor isolation valve will be designed to withstand this pressure and can be closed in ~2 min. During an actual fuel handling operation,

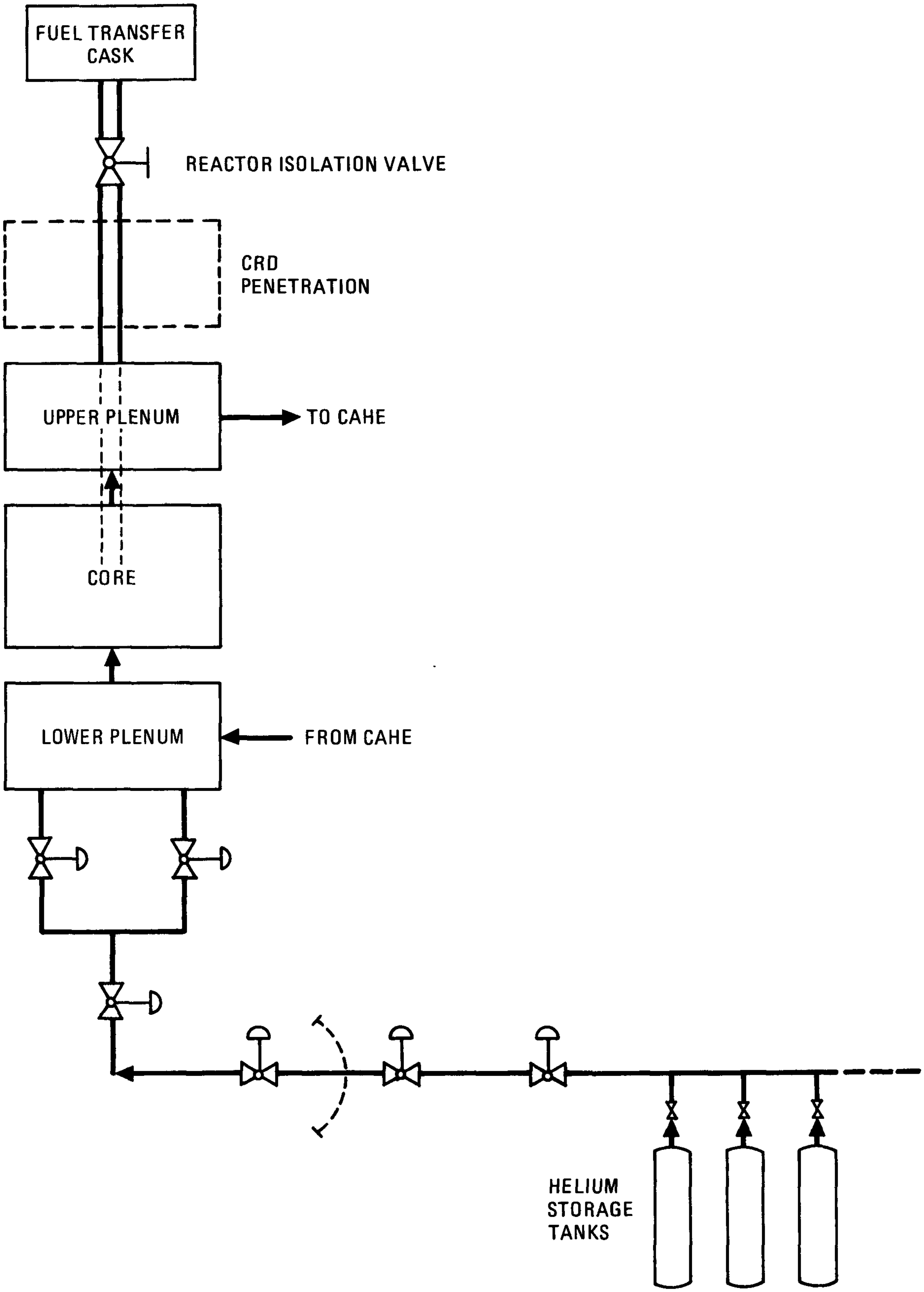

Fig. 4-28. Emergency repressurization

however, as much as a 15 min delay is possible in closing this valve. For this reason, the fuel transfer cask, which mates with the isolation valve, will also be designed to maintain this pressure.

4.5.4.3. Natural Circulation Initiation. No PPS action is required to initiate natural circulation on the CACS. The initiating event for natural circulation RHR is a total loss of drive power to the main circulators. The main circulators immediately begin coasting down, followed by a reactor scram due to a high power-to-flow ratio. Under such a scenario, the SCS pony motors would be expected to maintain forced circulation in the main loops. However, for one reason or another, the pony motors are assumed to be unavailable. In addition, an extremely conservative assumption is made that none of the three CACS circulators start. Fifty seconds after the loss of circulator drive power, core flow decays to less than 10%. The low flow then allows the MLIVs to shut by gravity, while the ALIVs fall open. The temperature profile existing in the auxiliary loop rapidly induces natural circulation between the core and the CAHEs. Simultaneously, the loss of power causes the CACS pump bypass check valve to open and the ALC louvers to completely open, both by gravity. The increased heat load to the CACS and reduced CACWS and ALC flow resistance increases natural circulation in the water and air loops.

The initiation of natural circulation during refueling is similar, except that action must be taken to close the reactor isolation valve and to open the helium storage valves to allow PCRV repressurization.

REFERENCES

4-1. The Institute of Electrical and Electronic Engineers, Inc. (IEEE) Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations."

4-2. The Institute of Electrical and Electronic Engineers, Inc., (IEEE) Standard 603-1980, "Criteria for Safety Systems for Nuclear Power Generating Stations."

4-3. "General Design Criteria for Nuclear Power Plants," in Code of Federal Regulations, Title 10, Part 50, Appendix A, U.S. Government Printing Office, Washington, D.C., 1977.

5. CORE COOLING SYSTEM PERFORMANCE*

5.1. GENERAL

To evaluate performance adequacy of the various core cooling features described in the previous sections, the following sections analyze the plant system response to transient events and summarize the results. The transient events which are particularly important in determining adequacy of the gas-cooled fast breeder reactor (GCFR) core cooling systems for residual heat removal (RHR) operations are selected from the following four event categories of Regulatory Guide 1.70 (Ref. 5-1):

1. Decrease in reactor primary coolant flow rate (Section 5.2).
2. Decrease in core heat removal by the secondary system (Section 5.3).
3. Decrease in reactor coolant inventory (Section 5.4).
4. Reactivity accidents (Section 5.5).

A spectrum of initiating events in each of the above categories is further classified into the five American Nuclear Society (ANS) plant conditions (PCs) that are defined according to the expected frequency of occurrence.

The adequacy of the GCFR RHR capability is examined with respect to the following:

1. Deterministic safety analysis rules (Section 5.1.1).
2. Performance margins in core cooling.
3. Redundancy margins in the backup system availability.

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*This section is presented to the NRC for information only and not for review and approval.

The GCFR RHR capability is determined by first ascertaining adequate core cooling during the accident scenarios, which meet the minimum requirements of the deterministic safety evaluation rules of ANS-50, Policy 2.4 (Ref. 5-2). The plant and RHR systems response to these deterministic accident scenarios (Section 5.1.1) indicates adequate core cooling with large RHR performance margins.

Results of the deterministic safety analyses also indicate that only part of the several redundant RHR systems is required to mitigate accidents. To demonstrate a large margin of GCFR system redundancy, margin cases are defined by assuming multiple failures beyond the deterministic analysis rules. These margin cases are summarized at the end of each subsection.

Cases of postulated loss of forced circulation (LOFC) and passive RHR by natural circulation are particularly important margin cases (see Section 5.6). Section 5.6 examines the case of LOFC under a station ac power loss for 2 h, an increasingly important licensing consideration, and examines accident mitigation using natural circulation during refueling.

The margin of core cooling performance shown in the results depends on the treatment of system parameter uncertainties in the analysis model. Most events are analyzed with a conservative model in which all the considered uncertainties (see Section 5.1.3.6) are applied cumulatively in the direction which is detrimental to core cooling. This treatment leads to very conservative results. To determine a realistic margin of safety, analysis of the most limiting case is expanded to include statistical treatment of the system uncertainties. Section 5.7.1 details this result for the DBDA and discusses the margin of RHR performance. Section 5.7.2 demonstrates the depth of redundancy margin provided by various backup RHR systems.

The following subsections summarize general data, conditions, and models applicable to all transient analyses. Sections on the specific event categories follow. Section 5.7 discusses RHR performance evaluation.

5.1.1. Selection of Transients and Accidents

The initial step in selecting transients and accidents is to establish a large spectrum of initiating events for the four categories identified above. The procedure of ANS-50, Policy 2.4 (Ref. 5-2) is then followed which provides for

1. A consistent application of postulated failures in plant systems and components responding to the initiating events.

2. A consistent application of safety criteria, in terms of radiological, system, or component design, or safety limits to the consequences of the initiating events or of the combined events (initiating event plus postulated failures).

This procedure represents a deterministic safety evaluation which can be applied at early stages in plant design when there is limited design definition and data base. As the design matures, a probabilistic assessment will provide additional design guidance.

The initial step in the ANS procedure is to classify the initiating events into five plant conditions (PCs) according to their expected frequency of occurrence during the life of the plant. Table 5-1 presents this classification system and defines the safety criteria for each PC. The safety objectives in relating design requirements to each PC are the following:

1. The most probable occurrences shall be accommodated by the largest design margin and yield the least radiological risk to the public.

2. Those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur.

Table 5-2 shows the approximate relationship of these PCs to existing NRC and ANS classification systems.

TABLE 5-1
EVENT CLASSIFICATION SYSTEM

Expected Best-Estimate Frequency of Occurrence (F) per Reactor Year	PC	Safety Criteria Off-site Dose Limit
Planned Operations	1	
$F \geq 10^{-1}$	2	Ref. 5-3,[a] 10CFR50, App. I
$10^{-1} > F \geq 10^{-2}$	3	1% Ref. 5-4,[b,c] 10CFR100, App. I
$10^{-2} > F \geq 10^{-4}$	4	10% Ref. 5-4,[b,c] 10CFR100, App. I
$10^{-4} > F \geq 10^{-6}$	5	100% Ref. 5-4,[b,c] 10CFR100, App. I

[a]The plant shall be designed to meet the dose objectives of 10CFR50, Appendix I (Ref. 5-3), for the summation of the radioactive releases due to PC-1 and the annual average radioactive releases (i.e., product of event release and event frequency) due to PC-2 events based on best estimated dose analyses.

[b]The plant shall be designed to meet the dose criteria for PC-3, PC-4, and PC-5 events based on conservative dose analyses, considering that initial plant parameters may be at their limiting values consistent with the Technical Specifications. Conservative dose analyses shall be performed using the methods which reasonably account for the actual plant and site characteristics and using methods which conservatively include the dose-reducing features of both the plant and the site.

[c]Augmented to include 75 rem lung and 150 rem bone. Prior to the issuance of a construction permit, the lower limits specified in Regulatory Guides 1.3 and 1.4 (Refs. 5-5 and 5-6), augmented to include 7.5 rem lung and 15 rem bone, shall apply.

TABLE 5-2
COMPARISON OF EVENT CLASSIFICATION SYSTEMS[a]

<table>
<tr><th rowspan="3">EVENT FREQUENCY RANGE (PER YEAR)</th><th rowspan="3">ANS-50[b] POLICY 2.4</th><th colspan="6">EXISTING TERMINOLOGY</th></tr>
<tr><th colspan="3">NRC</th><th colspan="3">ANS</th></tr>
<tr><th>10CFR[c]</th><th>RG 1.48,[d]</th><th>RG 1.70[e] REV.3</th><th>N18.2[f]</th><th>N212[g]</th><th>N213[h]</th></tr>
<tr><td>PLANNED OPERATIONS</td><td>PC-1</td><td>NORMAL</td><td>NORMAL</td><td>NORMAL</td><td>CONDITION I</td><td>NORMAL PPC[i]</td><td>PLANT CONDITION A</td></tr>
<tr><td rowspan="2"></td><td rowspan="2">PC-2</td><td rowspan="3">ANTICIPATED OPERATIONAL OCCURRENCES</td><td rowspan="3">UPSET</td><td>MODERATE FREQUENCY</td><td>CONDITION II</td><td rowspan="3">FREQUENT PPC</td><td rowspan="3">PLANT CONDITION B</td></tr>
<tr><td rowspan="4">INFREQUENT INCIDENTS</td><td rowspan="4">CONDITION III</td></tr>
<tr><td rowspan="2">10^{-1}</td><td rowspan="2">PC-3</td></tr>
<tr><td rowspan="8">ACCIDENTS</td><td rowspan="5">EMERGENCY</td><td rowspan="3">INFREQUENT PPC</td><td rowspan="4">PLANT CONDITION C</td></tr>
<tr><td rowspan="2">10^{-2}</td><td rowspan="4">PC-4</td></tr>
<tr><td rowspan="6">LIMITING FAULTS</td><td rowspan="6">CONDITION IV</td></tr>
<tr><td rowspan="2">10^{-3}</td><td rowspan="5">LIMITING PPC</td></tr>
<tr><td rowspan="4">PLANT CONDITION D</td></tr>
<tr><td>10^{-4}</td><td rowspan="2">PC-5</td><td rowspan="3">FAULTED</td></tr>
<tr><td>10^{-5}</td></tr>
<tr><td>10^{-6}</td><td>NOT CONSIDERED</td></tr>
</table>

[a] Boundaries shown by dashed lines are inferred.
[b] Ref. 5-2.
[c] Refs. 5-3 and 5-4.
[d] Ref. 5-7
[e] Ref. 5-1.
[f] Ref. 5-8.
[g] Ref. 5-9.
[h] Ref. 5-10.
[i] Plant process condition.

For each initiating event, failures in responding plant systems and components are postulated in accordance with the deterministic safety rules of ANS-50, Policy 2.4 (Ref. 5-2). These rules govern the combination of single failures and coincident occurrences with the initiating events. These rules are summarized below as they are applied to the GCFR:

1. Rule 1, failure classification

   a. Single failure. Applied to safety systems and their components.

      (1) Fluid systems.

         (a) Active. Applied to components requiring mechanical motion to perform safety function. Applicable at any time during plant response to the initiating event.

         (b) Passive. Applied to components not requiring mechanical motion to perform safety function. Applicable 24 h after initiating event.

      (2) Nonfluid systems. Applicable at any time during response to the initiating event.

   b. Coincident occurrence. Applied to nonsafety systems and components.

      (1) Probability of occurrence: (p) $\leq 10^{-2}$/event.
      (2) Probability of occurrence: (p) $> 10^{-2}$/event.

2. Rule 2, failure application

   Depending on the class of failure indicated in Rule 1, plant response within the radiological, design, or safety limits of the next higher PC is allowed when these failures are combined with the initiating events. This is analogous to the manner in which the Standard Review Plan (Ref. 5-11) for Chapter 15 of Safety Analysis Reports (SAR) allows increased consequences for initiating events combined with certain failures. Based on the PC of the initiating event (IE):

a. Single failure: $PC_{IE} + 1$ limits allowed (e.g., PC-3 + 1 → PC-4).

b. Coincident occurrence ($p \leq 10^{-2}$): $PC_{IE} + 1$ limits allowed.

c. Coincident occurrence ($p > 10^{-2}$): No increase in limits allowed.

d. Maximum limit change: $PC_{IE} + 2$, not to exceed PC-5 limits.

3. Rule 3, required postulated failures

   a. Single failure criterion

      (1) Emergency reactivity control. No failures were considered in this analysis.

      (2) Emergency core heat removal.

         (a) One shutdown cooling system (SCS) loop or core auxiliary cooling system (CACS) loop incapable of functioning when system is called upon.

         (b) One main loop cooling system (MLCS) loop helium isolation valve fails to close when loop is isolated (produces core bypass flow).

(3) Containment isolation. No failures were considered in this analysis.

b. Coincident occurrence: ($p \leq 10^{-2}$). If the turbine generator trips, a loss of off-site power (LOSP) is applied at the worst time during plant response.

c. Coincident occurrence: ($p > 10^{-2}$). Maximum worth control rod stuck in fully withdrawn position. Although this is a failure in a safety system, it is included in this class to reflect the requirements of the Standard Review Plan (Ref. 5-11) for Chapter 15 of SARs.

These deterministic safety rules were applied to the core cooling systems on a system-by-system and/or loop-by-loop basis according to the following sequence of actuation from the PPS: MLCS (three loops) → SCS (three loops) → CACS (three loops) → natural circulation CACS (three loops).

For each initiating event, all required postulated failures were applied in various combinations, with each combination reaching the maximum limit change (the lower of PC_{IE} + 2 or PC-5).

From this spectrum of combined events, both limiting and nonlimiting events for each of the above four categories were selected for analyses and are reported in the remainder of Section 5. Appendix A presents examples applying the ANS procedure.

5.1.2. Core and Essential Component Design Limits

The plant protection system (PPS) and the plant Technical Specifications are designed to insure that the plant operates within the specified design limits for the core and the essential components. To determine RHR adequacy, the temperature limits of these components are primarily addressed below.

Among the limit temperatures, the core fuel cladding temperature is the key to determine whether the RHR performance is adequate in all the undercooling transient cases, which are presented in the later sections. The maximum fuel centerline temperature is important for the overpower transient in reactivity accidents (such as inadvertent control rod withdrawal).

5.1.2.1. Core Assembly Temperature Design Limits. The GCFR fuel and blanket rods contain uranium and plutonium mixed oxide fuel pellets in stainless steel cladding. As in the light water reactor (LWR) fuel, the design limit temperature for the oxide fuel pellet is defined to be the oxide melting temperature at the hot spot rod center. The oxide melting temperature has a range. The lower bound value, 2650°C (4800°F), is used for the normal operation limit (PC-1), and phase change temperature from solidus to liquidus of 2800°C (5070°F) is used for all accident conditions (PC-2 through PC-5) (see Table 5-3).

Reference 5-12 defines the limiting temperature at the faulted condition (i.e., the ANS PC-5) for the stainless steel cladding for the GCFR fuel assembly as 1300°C (2370°F). This temperature is determined to be a conservative value after examining physical parameters affecting the integrity of the GCFR fuel assembly (Ref. 5-12). This temperature is conservative because, based on the evaluation of the available data concerning strength, melting, and oxidation of stainless steel type 316, loss of coolable core geometry in a GCFR fuel assembly is not expected to occur below 1370°C ± 25°C (2500°F ± 50°F). The GCFR cladding faulted temperature of 1300°C (2372°F) is also conservative with respect to the faulted limits of other reactors using stainless steel cladding [e.g., 1370°C ± 50°C (2500°F ± 90°F) for the Clinch River Breeder Reactor (CRBR) (Ref. 5-13), 1350°C (2462°F) for the advanced gas-cooled reactor (AGR) (Ref. 5-14), and 1300°C (2372°F) for the gas-cooled reactor (GBR) (Ref. 5-15)]. The faulted temperature will be experimentally verified. Reference 5-12 details the test programs planned for this purpose, the basis of the current selection for the faulted limit, and a review of physical parameters affecting the integrity of the GCFR fuel assembly. The NRC review and concurrence of the fuel cladding limits was

TABLE 5-3
CORE AND BLANKET ROD CLADDING AND FUEL DESIGN TEMPERATURE LIMITS

ANS PC	Frequency of Occurrence Per Year (F)	Cladding [°C (°F)]	Fuel [°C (°F)]
1	Normal operation	750 (1382)	2650 (4800)
2	F > 10-1	850 (1562)	2800[a] (5070)
3	$10^{-1} \geq F > 10^{-2}$	950 (1742)	2800 (5070)
4	$10^{-2} \geq F > 10^{-4}$	1100 (2012)	2800 (5070)
5	$10^{-4} \geq F > 10^{-6}$	1300 (2372)	2800 (5070)

[a] No centerline melting.

requested when Ref. 5-12 was submitted and is not requested as part of the review of this report on RHR systems.

For other lower PCs, lower cladding temperatures are defined to account for higher probability of occurrence (see Table 5-3).

In accident cases of very high cladding temperatures in the neighborhood of the PC-5 cladding limit (typically in a DBDA case), an allowance must be made to acccount for a local excess temperature occurring at the peripheral rods in the fuel assembly edge adjacent to the duct wall. This edge channel overheating effect is due to (1) gamma heating in the duct wall, which becomes a significant heat source after a reactor trip, and (2) the flow redistribution in favor of the bundle interior subchannels after laminar flow transition. The effect of the roughened rod surface on the friction vanishes in laminar flow. This effect is less pronounced in the peripheral subchannels since the duct wall is not roughened. Consequently, the coolant tends to be diverted away from the edge subchannels and into the bundle interiors under a laminar flow regime.

Since the circumferential and transverse conductions within the fuel rods and asymmetrical radiation are important in the edge subchannels, detailed two-dimensional analyses were performed for a pseudo-steady-state condition at the time of the peak temperature. The edge rod cladding temperature is up to 100°C (180°F) higher than that for the typical interior rods. Appendix B.1 discusses the two-dimensional analysis in more detail.

All the system dynamics analyses and the results for the transient fuel cladding response, presented in the Section 5 figures, are based on the one-dimensional model of the typical interior fuel rods; however, the most limiting DBDA case shows a 200°C margin to the PC-5 cladding temperature, which is adequate to offset the edge channel overheating effect. For PC-4 and lower PC cases, the edge channel overheating effect is negligible because the core flow is typically turbulent and subchannel enthalpy mixing is significant.

In the blanket assemblies, the edge channel overheating does not occur, because the edge subchannel adjacent to the duct wall has a relatively large flow area and is always overcooled before and after a reactor trip. However, due to a large radial power gradient in the blanket assembly, a detailed two-dimensional analysis is necessary and was performed based on the typical pseudo-steady-state at the time of the maximum cladding temperature. Appendix B.2 summarizes this analysis in more detail. All the transient analyses and the results for the blanket maximum temperatures in the Section 5 figures are based on a one-dimensional model of the blanket rod that approximates the hot spot blanket cladding temperature determined by the two-dimensional analysis.

Thermal response of the blanket rod is significantly slower than the fuel rod due to its larger rod diameter. Transient analyses generally show that the blanket rod cladding temperature limits are met so long as the fuel cladding temperature meets its design limit.

5.1.2.2. Essential Component Design Limits. Table 5-4 shows the design limit temperatures for the essential primary loop components, other than the core, determined for the various PCs, for currently selected materials, and for the expected number of thermal transient cycles during the plant life. The PPS and the plant Technical Specifications will insure that these components are not to be exposed to temperatures higher than these limits during any steady-state or transient operation.

To determine whether adequate RHR is provided, the transient temperatures of the primary coolant at the component location are examined with respect to the component design limits, conservatively neglecting the temperature attenuation in the coolant boundary layers. In particular, the reactor mixed outlet temperature is compared with the limit temperatures of the Class B thermal barrier and the steam generator, while the reactor inlet temperature is compared with the limits for the Class A thermal barrier and the circulator. In some transient cases, a hot coolant streak temperature from the central core is higher than the Class B thermal barrier limit. A

TABLE 5-4
ESSENTIAL COMPONENT DESIGN TEMPERATURE LIMITS

ANS PC	Frequency of Occurrence Per Year (F)	Circulators [°C (°F)]	Steam Generator [°C (°F)]	Thermal Barrier[a] Class A (Cold Region) [°C (°F)]	Thermal Barrier[a] Class B (Hot Region) [°C (°F)]
1	Normal Operation	315 (600)	550 (1022)	315 (600)	550 (1022)
2	$F > 10^{-1}$	345 (653)	650 (1202)	400 (752)	630 (1166)
3	$10^{-1} \geq F > 10^{-2}$	345 (653)	650 (1202)	400 (752)	630 (1166)
4	$10^{-2} \geq F > 10^{-4}$	427 (800)	790 (1454)	565 (1050)	980 (1800)
5	$10^{-4} \geq F > 10^{-6}$	427 (800)	790 (1454)	565 (1050)	980 (1800)

[a] Both the Class A and B thermal barriers use stainless steel 316, but they have different types of insulation suitable for their service temperatures.

detailed analysis has indicated that the hot streak temperature attenuates substantially and that the thermal barrier temperature remains well below the design limit.

A summary of transient analyses for various cases in Table 5-5 shows that the primary coolant temperatures adequately meet the component design limits of Table 5-4 as long as the core temperature limits are met.

5.1.3. Plant Characteristics Considered in the Cooling System Performance Evaluation

5.1.3.1. Plant Design Conditions. The plant is designed to generate rated nuclear steam supply system (NSSS) thermal output. This power output includes the thermal power generated by the primary coolant circulators and is consistent with the license application rating.

The control system automatically maintains prescribed conditions, including the rated NSSS thermal output in the plant even under a conservative set of system stability and transient performance reactivity parameters. For each mode of plant operation, controller setpoints are derived and determined to satisfy plant operational requirements throughout the core life and for various power levels.

In accident analyses for review by the NRC, the initial power operating conditions are usually based on the proposed licensed core power level and an error allowance in steady-state power determinations.

5.1.3.2. Initial Conditions. For conservative accident evaluation, initial conditions are obtained by adding the maximum steady-state errors to rated values. The following steady-state errors are considered:

1. Core power: ±2% allowance for calorimetric error.
2. Average pressure: ±207 kPa (±30 psi) allowance for steady-state fluctuations and measurement error.

TABLE 5-5
SUMMARY OF RHR PERFORMANCE ANALYSIS, COMPARISON OF DESIGN LIMITS AND RESULTS OF ANALYSIS AND NOTATION
FIGS. 5-40 AND 5-41[(a)]

Symbol	Initiating Event	Number of Additional Failures	Type of Failures	Applicable Plant Condition	Ultimate Cooling Systems	Peak Cladding Temp (°C) Calculated Fuel	Peak Cladding Temp (°C) Calculated Blanket	Peak Cladding Temp (°C) Limit	Peak Fuel Temp (°C) Calculated	Peak Fuel Temp (°C) Limit	Peak Reactor Outlet Plenum Temp (°C)	Thermal Barrier Limit Temp (°C)
	<u>Decrease in Reactor Coolant Flow</u>											
△	Loop trip without power run-back	1	1 MLCS	PC-2	2 MLCS	826	804	850	2304	2800	541	630
□	Loss of all circulator power + LOSP	—	0	PC-3	3 SCS	758	791	950	2537	2800	523	630
◇	Loss of all circulator power + LOSP	1	1 SCS	PC-4	2 SCS	832	888	1100	2537	2800	524	980
▽	Circulator bearing seizure + LOSP	0	--	PC-4	2 SCS	890	918	1100	2537	2800	544	980
○	Circulating bearing seizure + LOSP	1	MLIV	PC-5	2 SCS	856	920	1300	2537	2800	524	980
⍏	Circulator bearing seizure + LOSP	1	1 SCS	PC-5	1 SCS	890	932	1300	2537	2800	545	980
▲	Circulator bearing seizure + LOSP	2	2 SCS	Beyond PC-5	3 CACS	888	933	1300	2537	2800	544	980
■	Circulator bearing seizure + LOSP	3	MLIV/ 2 SCS	Beyond PC-5	3 CACS	888	934	1300	2537	2800	544	980

TABLE 5-5 (Continued)

Symbol	Initiating Event	Number of Additional Failures	Type of Failures	Applicable Plant Condition	Ultimate Cooling Systems	Peak Cladding Temp (°C) Calculated Fuel	Peak Cladding Temp (°C) Calculated Blanket	Peak Cladding Temp (°C) Limit	Peak Fuel Temp (°C) Calculated	Peak Fuel Temp (°C) Limit	Peak Reactor Outlet Plenum Temp (°C)	Thermal Barrier Limit Temp (°C)
◆	Circulator bearing seizure + LOSP	3	2 SCS/ 1 CACS	Beyond PC-5	2 CACS	888	936	1300	2537	2800	544	980
	Decrease in RHR by Secondary System											
⌽	Loss of all feedwater	1	1 SCS	PC-4	2 SCS	807	871	1100	2538	2800	525	980
▼	Loss of condenser vacuum	3	3 SCS	Beyond PC-5	3 CACS	807	876	1300	2541	2800	525	980
●	Loss of condenser vacuum	4	3 SCS/ 1 CACS	Beyond PC-5	2 CACS	807	878	1300	2538	2800	525	980
▲	Loss of condenser vacuum	4	3 SCS/ 1 MLIV	Beyond PC-5	3 CACS	807	878	1300	2538	2800	525	980
■	Loss of condenser vacuum	5	3 SCS/ 2 CACS	Beyond PC-5	1 CACS	807	894	1300	2538	2800	525	980
◆	Loss of condenser vacuum	5	3 SCS/ 1 MLIV/ 1 CACS	Beyond PC-5	2 CACS	807	882	1300	2538	2800	525	980

TABLE 5-5 (Continued)

Symbol	Initiating Event	Number of Additional Failures	Type of Failures	Applicable Plant Condition	Ultimate Cooling Systems	Peak Cladding Temp (°C) Calculated Fuel	Peak Cladding Temp (°C) Calculated Blanket	Peak Cladding Temp (°C) Limit	Peak Fuel Temp (°C) Calculated	Peak Fuel Temp (°C) Limit	Peak Reactor Outlet Plenum Temp (°C)	Thermal Barrier Limit Temp (°C)
	<u>Decrease in Reactor Coolant Inventory</u>											
	DBDA	0	--	PC-5	3 MLCS	729	828	1300	2589	2800	556	980
	DBDA	1	1 MLCS	PC-5	2 MLCS	898	899	1300	2536	2800	570	980
	DBDA	2	1 MLCS/ 1 MLIV	PC-5	2 MLCS	1086	1100	1300	2536	2800	496	980
	DBDA	2	2 MLCS	Beyond PC-5	1 MLCS	867	975	1300	2299	2800	573	980
	DBDA with a large leak area	1	No flow restric- tor	Beyond PC-5	3 MLCS	917	1047	1300	2549	2800	554	980
	DBDA + LOSP	4	3 MLCS/ 1 CACS	PC-5	2 CACS	1097	1196	1300	2549	2800	565	980
	DBDA + LOSP	4	3 MLCS/ 1 MLIV	PC-5	3 CACS	826	1093	1300	2543	2800	580	980
	DBDA + LOSP NC in CACWS	6	3 MLCS/ 1 CACS/ 2 CACWS	Beyond PC-5	2 CACS	1127	1260	1300	2558	2800	576	980

TABLE 5-5 (Continued)

Symbol	Initiating Event	Number of Additional Failures	Type of Failures	Applicable Plant Condition	Ultimate Cooling Systems	Peak Cladding Temp (°C) Calculated Fuel	Peak Cladding Temp (°C) Calculated Blanket	Peak Cladding Temp (°C) Limit	Peak Fuel Temp (°C) Calculated	Peak Fuel Temp (°C) Limit	Peak Reactor Outlet Plenum Temp (°C)	Thermal Barrier Limit Temp (°C)
	Reactivity Accidents											
	Control (rod withdrawal) at 100% power	1	--	PC-4	3 SCS	874	895	1100	2800	2800	556	630
	Control (rod withdrawal) at 100% power	2	1 SCS	PC-5	2 SCS	874	895	1300	2800	2800	556	630
	Control rod withdrawal at 30% power	1	--	PC-4	3 SCS	1025	877	1100	1745	2800	592	980
	Control rod withdrawal at 30% power	2	1 SCS	PC-5	2 SCS	1025	877	1300	1745	2800	592	985
	LOFC											
	Loss of all ac power	0	--	Beyond PC-5	3 NC	837	945	1300	2536	2800	531	980
	Loss of all ac power	1	1 NC	Beyond PC-5	2 NC	837	972	1300	2536	2800	531	980
	LOFC during refueling with repressurization	1	1 NC	Beyond PC-5	2 NC	1032	1212	1300	1212	2800	469	980

(a)Legend:

MLCS = main loop cooling system
CACS = core auxiliary cooling system
PC = ANS plant condition
MLIV = main loop isolation valve
LOSP = loss of off-site power
SCS = shutdown cooling system
NC = natural circulation
DBDA = design basis depressurization accident

Initial values for core power, average coolant temperature, and pressure are selected to minimize the core thermal margin unless stated otherwise in the sections describing specific accidents. Table 5-6 summarizes initial conditions for the best estimate and conservative analyses.

5.1.3.3. Power Distribution in Core. The transient response of the core is dependent on the initial power distribution. The power distribution may be characterized by the radial and the axial peaking factors.

The transient analyses are based on the maximum peaking factors determined by the nuclear design. The axial power shape for the fuel rod used in the transient analysis is a chopped cosine, as shown in Fig. 5-1. The axial peaking factor is 1.25. Figure 5-2 shows the axial power profile for the radial blanket assembly used for the transient analysis. The reactor coolant system transient response is based on the core with the average powered assemblies. Since the limiting thermal conditions occur in the maximum powered fuel rod and in the maximum powered blanket rod in some cases, these rods are analyzed with and without hot spot engineering factors. Section 5.1.3.6 details the hot spot engineering factors. The maximum power conditions in the fuel and in the blanket are based on the beginning-of-life and the end-of-life exposures, respectively, since the fuel assembly power decreases as exposure increases while the blanket assembly power increases as exposure accumulates.

In calculating power generated in the fuel or blanket rods, the fraction of power generated in the structures, such as the cladding and the duct wall, is conservatively assumed to be included in the power generation calculated for the fuel or blanket materials. However, in special analyses for the detailed local thermal-hydraulic effects, such as the fuel assembly edge channel and the hot channel at the core-blanket interface, where gamma heating in the assembly duct material is important, the structural gamma heating is accounted for separately.

TABLE 5-6
PLANT INITIAL CONDITIONS ASSUMED FOR ACCIDENT ANALYSES

	Best-Estimate Analysis	Conservative Analysis
Reactor power [MW(t)]	1088	1110
Reactor coolant mass flow rate [kg/s (lb.s]		
Reactor coolant pressure (MPa (psia)]	10.4 (1508)	10.5 (1518)
Core inlet temperature [°C (°F)]	296 (564)	301 (574)
Core outlet temperature °°C (°F)]	520 (968)	524 (975)
Circulator speed (rpm)	2830	3090

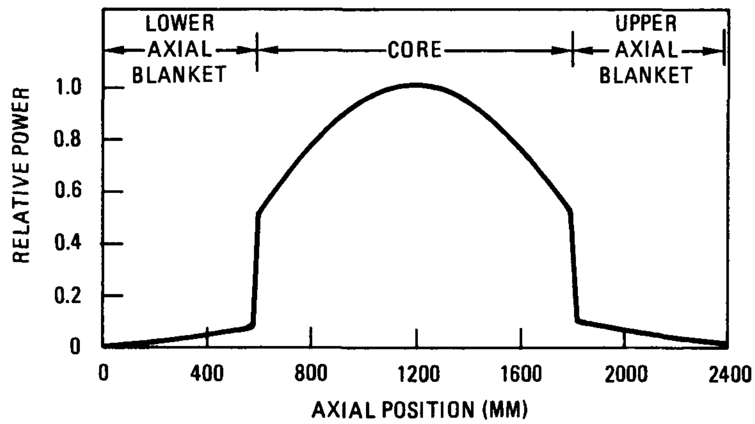

Fig. 5-1. Relative axial power distribution in average fuel rod assembly

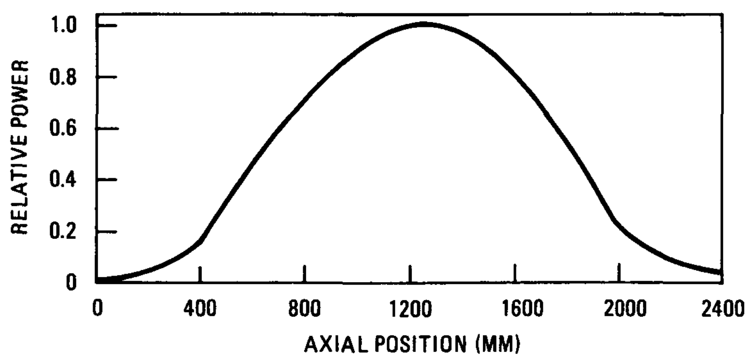

Fig. 5-2. Relative axial power distribution in average radial blanket assembly

5.1.3.4. Reactivity Coefficients Assumed in the Accident Analysis. The transient response is dependent on reactivity feedback effects, particularly the Doppler power coefficient and the core axial expansion coefficient, which Ref. 5-16 details. In the analysis of certain events, conservatism requires the use of large reactivity coefficient values, whereas, in the analysis of other events, conservatism requires the use of small reactivity coefficient values. However, the variation to obtain conservatism has little effect in most RHR analyses. Therefore, a set of best estimate reactivity coefficients, listed in Table 5-7, are uniformly assumed for all the transient analyses.

5.1.3.5. Control Rod Insertion Characteristics. The negative reactivity insertion following a reactor trip is a function of the acceleration of the rod absorber, and the variation in rod worth is a function of rod position. With respect to accident analyses, the critical parameter is the time of full insertion or ~85% of the rod cluster travel. The GCFR is equipped with two sets of absorbers used for reactor trip: (1) control rods, used for reactivity control, and (2) shutdown rods, used for the secondary shutdown. The control rods are inserted rapidly by the action of spring-assisted gravity. The shutdown rods are also inserted by gravity from the fully out to the fully inserted position. The shutdown rod insertion is backed up by the motor drive with a delay.

Figure 5-3 shows the control and shutdown rod positions versus time assumed in the accident analyses. Figure 5-4 shows the fraction of total negative reactivity insertion versus normalized rod position for the control or shutdown rods.

Figures 5-5 and 5-6 show the relative negative reactivities inserted by control and shutdown rods following a reactor trip, respectively. These curves, which were obtained from Figs. 5-3 and 5-4, are used in conjunction with a point kinetics core model for all the transient analyses.

5.1.3.6. System Parameter Uncertainties. A conservative model for accident analyses was defined by incorporating uncertainty margins for the parameters

TABLE 5-7
REACTIVITY FEEDBACK COEFFICIENTS USED FOR ACCIDENT ANALYSES

$\beta_{effective}$	0.003945	
Doppler	$-0.0032\ \Delta P/(\Delta T/T)$	$= -0.81\ \$/(\Delta T/T)$
Axial expansion	$-0.0338\ \$/mm$	$= -10.30\ \$/ft$
Helium density	$-40.21\ \$/(kg/m^3)$	$= -2.51\ \$/(lb/ft^3)$
Element bowing[a]	0.0	
Grid plate bending[a]	0.0	
Grid plate expansion[a]	0.0	

[a] Zero values were used because the values were not available at the time and because they are insignificant in comparison with the rod insertion effects.

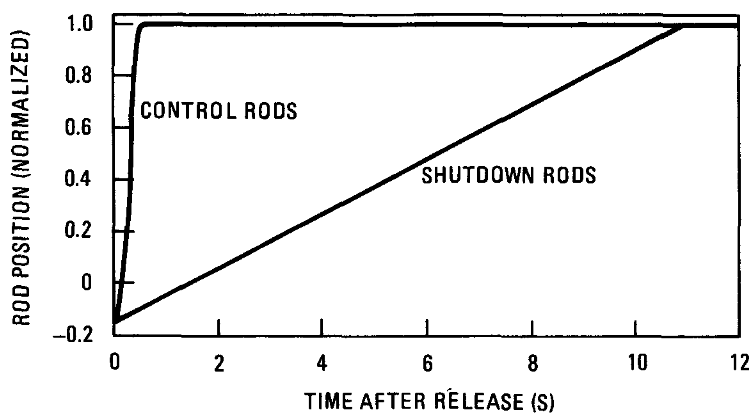

Fig. 5-3. Normalized rod position versus time after rod release for control and shutdown rods

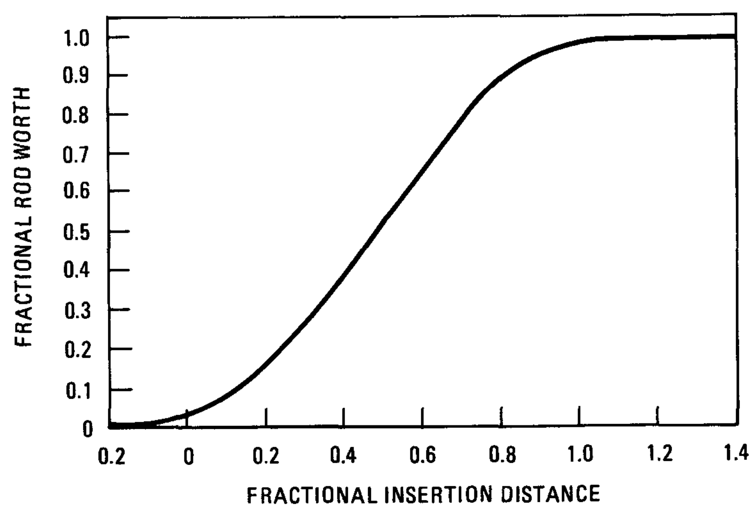

Fig. 5-4. Normalized rod worth for control and shutdown rods

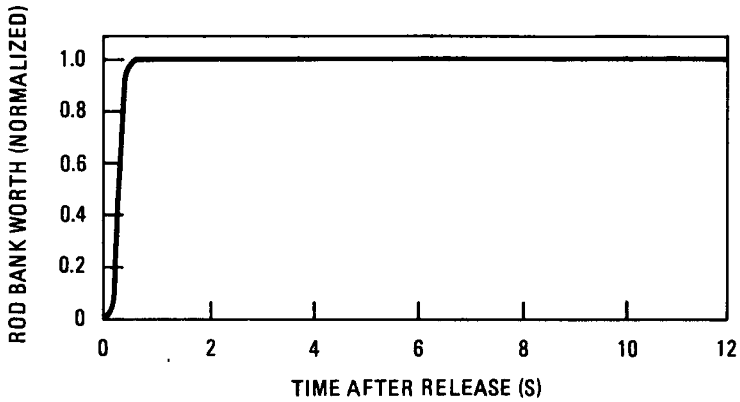

Fig. 5-5. Relative negative reactivity inserted by control rod bank versus time

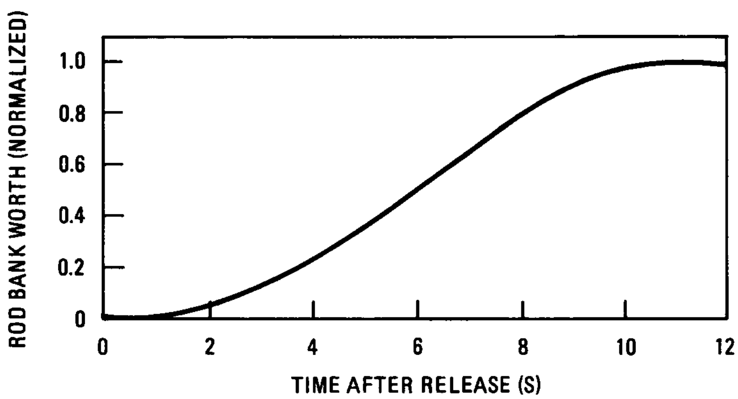

Fig. 5-6. Relative negative reactivity inserted by shutdown rod bank versus time

related to the GCFR system performance. The system uncertainty values were obtained primarily from similar studies done for the high-temperature gas-cooled reactor (HTGR), using values specific to the GCFR where appropriate. In the conservative model, each uncertainty factor is assumed to be in its most detrimental direction for core cooling.

To determine a realistic margin of safety, analysis of the most limiting case is expanded to include a best-estimate model without the uncertainties and a statistical model with a statistical treatment of the uncertainties (Section 5.7.1.).

Table 5-8 summarizes the system uncertainty factors used to define the conservative and best estimate models. Reference 5-17 discusses each of these factors, except the fuel and blanket rod hot spot factors.

The fuel and blanket rod factors are defined to represent the effects of various uncertainties on the hot spot temperatures in the fuel or the blanket rods. These are the following: channel factors, which summarize the effect on the the coolant channel enthalpy rise; film factor, which accounts for the uncertainty effect in the surface heat transfer coefficient; and cladding factor, which allows for uncertainty in the cladding thermal conductance. The effects of uncertainties of statistical nature, such as physical property correlations and engineering tolerances in the fuel rod geometry, are combined statistically, and those of nonrandom nature, such as physics methods, are combined cumulatively.

The hot spot factors based on two or three standard deviations are used for best-estimate or conservative analyses, respectively, as indicated in Table 5-8.

For the steady state prior to all transients, the conservative and best-estimate initial conditions specified in Section 5.1.3.2 are assumed in addition to the uncertainty factors in Table 5-8.

TABLE 5-8
SYSTEM UNCERTAINTY FACTORS USED FOR BEST ESTIMATE AND CONSERVATIVE MODELS

Parameter	Best-Estimate Model		Conservative Model	
Decay heat	1.0		1.2	
Initial Power	1.0		1.02	
Local Power	1.0		1.05	
Overall conductance of CACS heat exchangers [CAHE and auxiliary loop cooler (ALC]	1.0		0.8075	
Flow pressure drops				
Primary helium loop	1.0		1.2	
Secondary water loop	1.0		1.2	
Tertiary air flow	1.0		1.2	
Coolant thermal conductivity	1.0		1.928	
Coolant viscosity	1.0		1.045	
Containment absolute back-pressure (DBDA only)	1.0		0.83	
Core bypass flow fraction (nominally 3.4%)	1.0		2.0	
Fuel Cladding Engineering Factors[a]				
Flow regimes	Turbulent	Laminar	Turbulent	Laminar
Hot channel factor	1.111	1.187	1.131	1.236
Hot film factor	1.311	1.173	1.215	1.212
Hot cladding factor	1.153	1.116	1.195	1.138
Hot fuel factor	1.142	1.142	1.178	1.178
Blanket Cladding Engineering Factors				
Hot channel factor	1.151	1.249	1.196	1.332
Hot film factor	1.710	1.395	1.788	1.477
Hot cladding factor	1.201	1.128	1.252	1.142
Hot fuel factor	1.175	1.175	1.212	1.212

[a]Hot spot factors for the best estimate and conservative models are based on two and three standard deviations, respectively.

5.1.4. Residual Decay Heat

Residual heat in a subcritical core is calculated in accordance with the requirements of 10CFR50, Appendix K (Ref. 5-18), which includes an assumption of infinite irradiation time before reactor trip under accident condition. Important decay heat correlations in the GCFR are those for the fuel rods, the blanket rod at the core-blanket interface, and the gamma heating in the structural members, such as the fuel assembly duct wall at the core center. Due to different decay characteristics, these correlations are defined separately, as described in the following sections. An uncertainty margin of 20% (see Table 5-8) for the first 1000 s and 10% thereafter was assumed according to the ANS-5 subcommittee recommendation (Ref. 5-19). This uncertainty margin is assumed for all the three decay heat correlations for the fuel, the blanket, and the structural gamma.

5.1.4.1. Core Fuel Decay Heat. The fission and breeding product decay heat production rates vary with time after reactor trip as

$$Q(t)/Q(0) = a \cdot t^{-b} + c \cdot e^{-t/\tau_U} + d \cdot e^{-t/\tau_{Np}} \quad ,$$

where $Q(t)/Q(0)$ = local power production expressed as a fraction of the full power value,

t = time after reactor trip, and

a and b = coefficients depending upon the time after reactor trip as defined by Shure (Ref. 5-19) as

t (s)	a	b
0 - 0.1	0.07	0
0.1 - 10	0.0603	0.0639
10 - 150	0.0765	0.1807
150 - 4×10^6	0.1301	0.2834

The latter two terms in the above equation account for the decay of the U-239 and Np-239 breeding products, respectively; τ_U and τ_{Np} are the (exponential) decay times of the two uranium and neptunium isotopes equal to 2034 and 292,400 s, respectively; the coefficients c and d relate to the energy production rates associated with each isotope and are equal to 0.00129 and 0.00112, respectively. Interestingly, the breeding product decay heat production rate is much lower than the fission product rate for short times after reactor trip, but they become quite significant for long times after shutdown. The power production due to delayed neutron fission is also included in the local power production evaluation by solving the six delayed neutron group point kinetics equations. Figure 5-7 shows the normalized fuel rod decay heat generation rate with time.

5.1.4.2. Blanket Decay Heat. Ascertaining adequate cooling of the radial blanket rods is important, since these rod temperatures can be limiting under accident conditions. In the radial blanket, especially the first row rods, gamma energy transported from the core is much greater than gamma energy transported out of the blanket. Figure 5-8a shows the decay power radial distribution in the blanket assemblies.

Because of different decay characteristics due to breeding product and gamma energy transport from the core, a decay heat correlation was developed (Ref. 5-20) specifically for the blanket rods. Figure 5-8b gives the normalized decay power variation with time for just two rows of blanket rods. This temporal variation and the spatial distribution discussed above was used for analyzing thermal response of the maximum powered blanket rod under all accident conditions.

5.1.4.3. Gamma Heating in the Core Central Structures. A decay heat correlation for steel components at the core center was determined and used for analyzing the detailed thermal-hydraulic effect in the fuel assembly edge subchannels which are formed adjacent to the duct wall. The duct wall gamma heating becomes a relatively significant heat source for the edge subchannels at accident conditions, which are detailed in Appendix B.

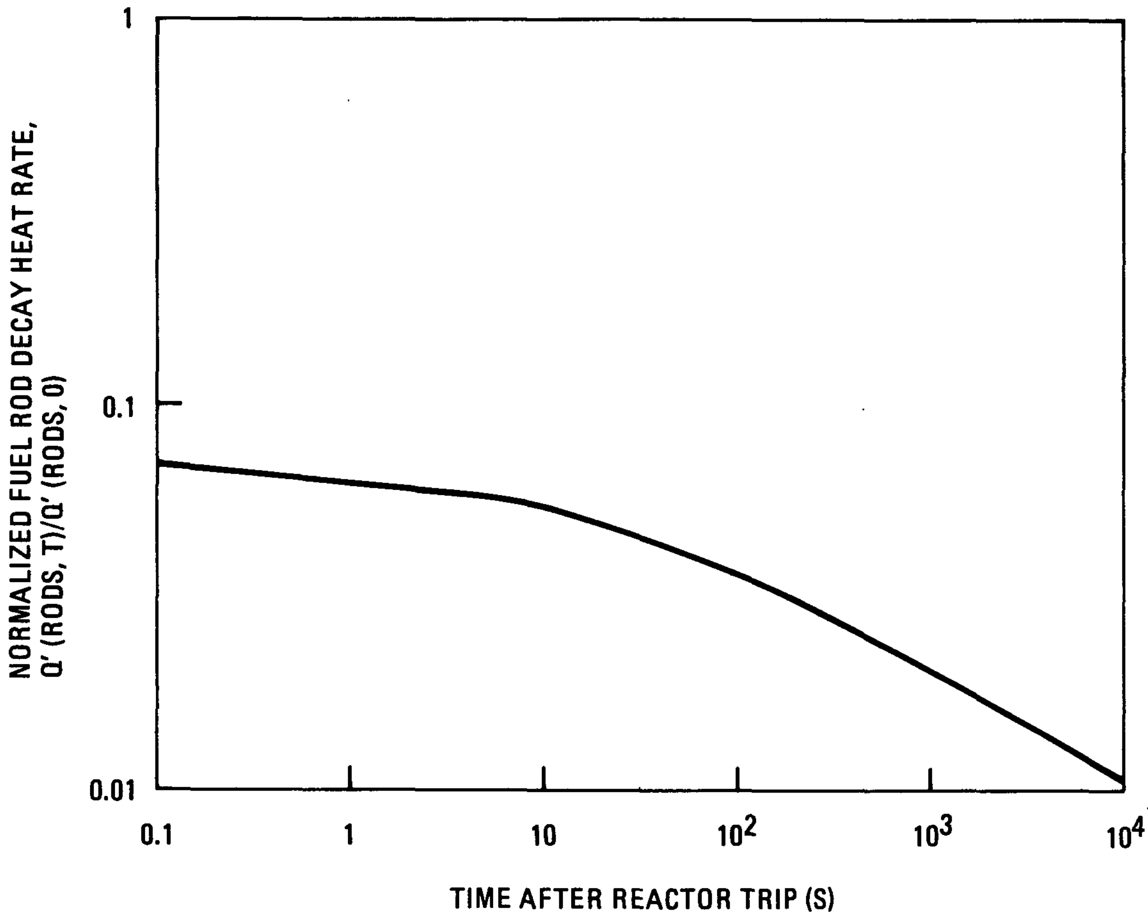

Fig. 5-7. Normalized fuel rod decay heat rate

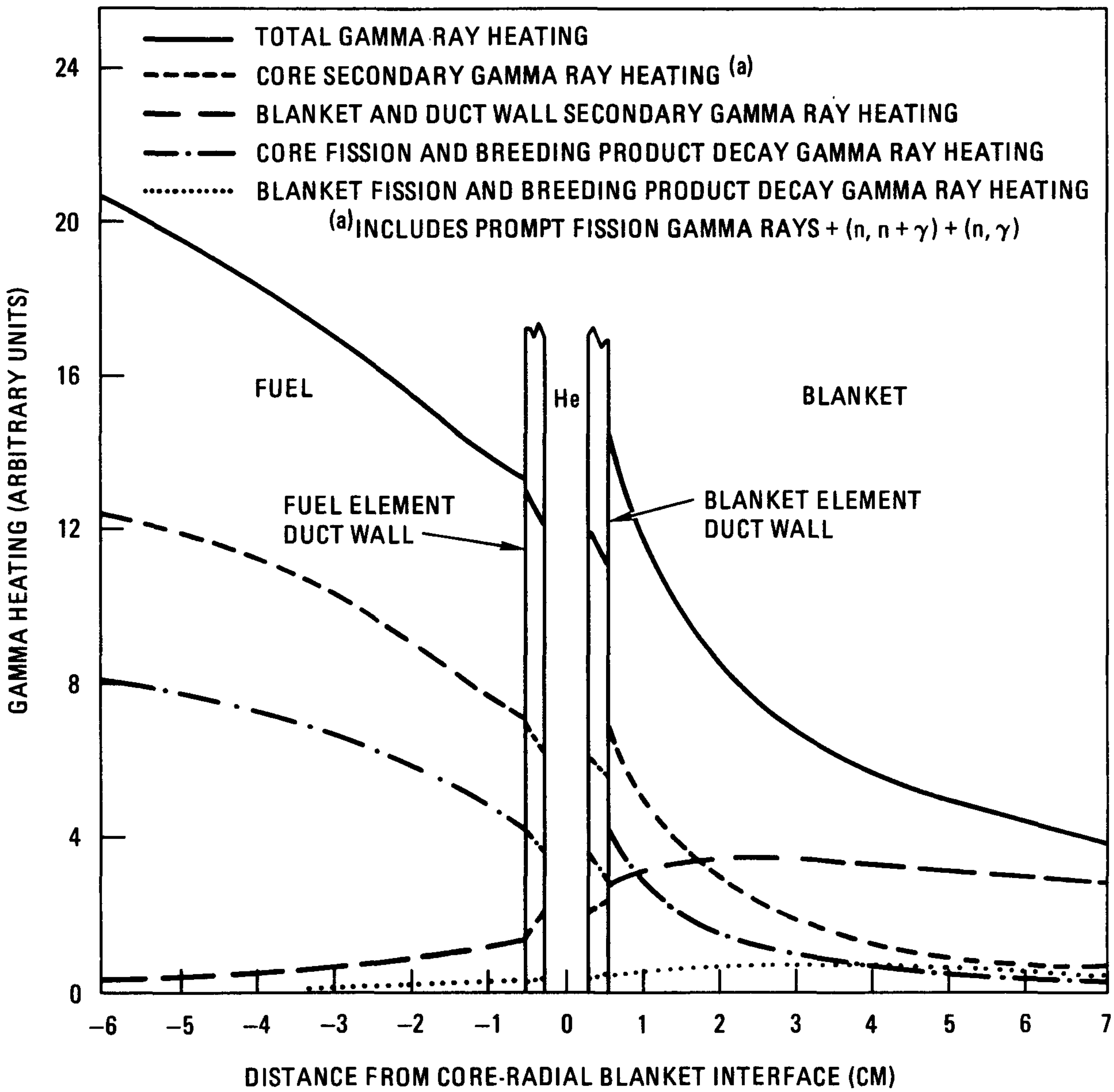

Fig. 5-8. Radial blanket decay heat: (a) gamma heating distribution at the core/radial blanket interface in the GCFR demonstration plant (sheet 1 of 2)

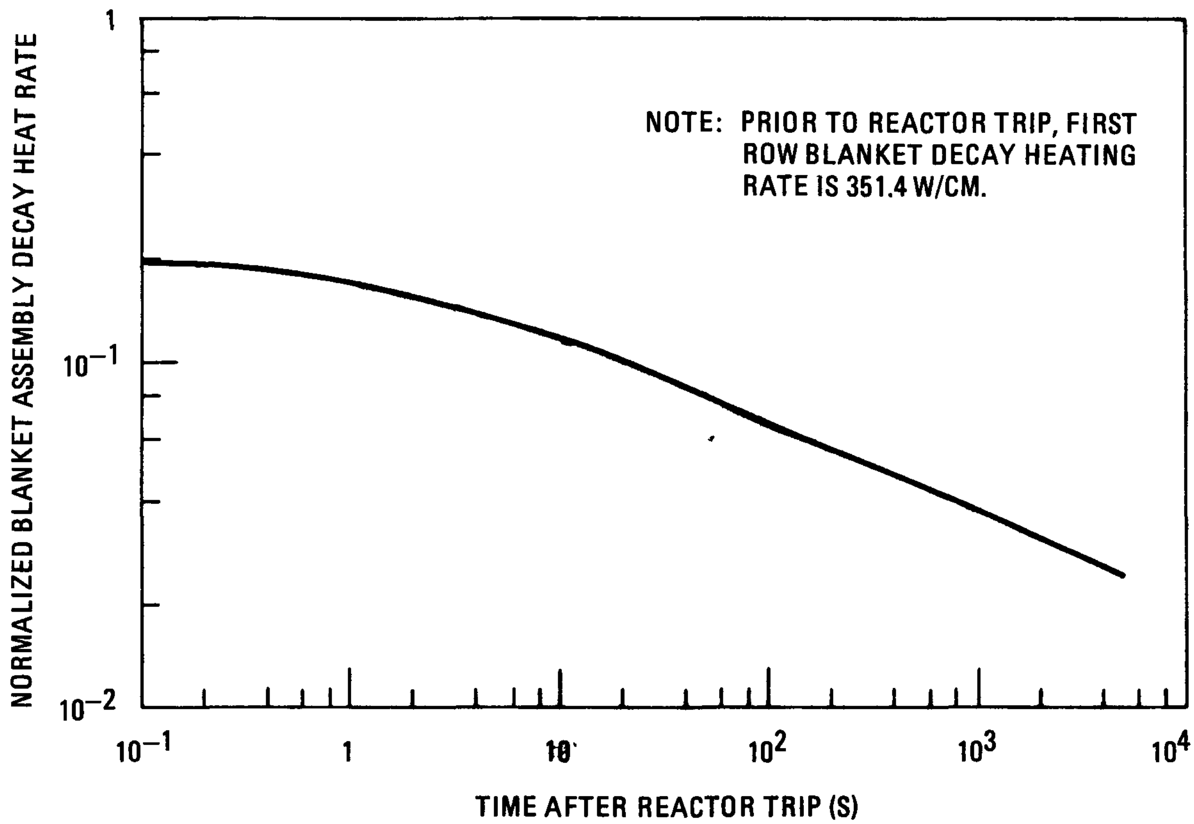

Fig. 5-8. Radial blanket decay heat: (b) normalized decay heat in radial blanket (sheet 2 of 2)

The total steel heating rate is the sum of heating rates due to fission events and to fission and breeding product decay gammas. The analysis is based on Ref. 5-20. Figure 5-9 shows the ratio of duct wall heating rate to the fuel rod heating rate at time t.

5.1.5. Assumed Protection System Actions

The GCFR is designed to properly protect against the possible effects of natural phenomena, postulated environmental conditions, and postulated accidents. The quality assurance program will be implemented to assure that the plant will be designed, constructed, and operated without undue risk to the health and safety of the general public. The incorporation of these features, coupled with the reliability of the design, ensures that the normally operating systems and the engineered safety feature system will mitigate the events discussed herein.

The GCFR PPS is comprised of reactor trip systems and RHR initiation and termination systems, which are detailed in Sections 4.4, 4.5.1.3, 4.5.2.3, and 4.5.3.3. The following sections summarize the actions assumed for analysis of the core cooling system performance, including delay times.

5.1.5.1. Reactor Trip System. The GCFR PPS contains two diverse and redundant reactor trip systems, the primary and secondary reactor trip systems. Each trip system has an independent and diverse logic system. Additionally, the PPS contains the RHR initiation and termination systems. The following sections summarize the simulation assumption for the reactor trip systems and the analysis of the core cooling system performance.

Primary Reactor Trip System. The primary trip system releases the gravity-actuated control rods, using the signals from the primary trip parameters (Fig. 4-13). Section 4.4.1 details the primary trip actuation.

Each trip input has various associated instrumentation delays, including delays in signal acquisition, in opening the trip breakers, and in

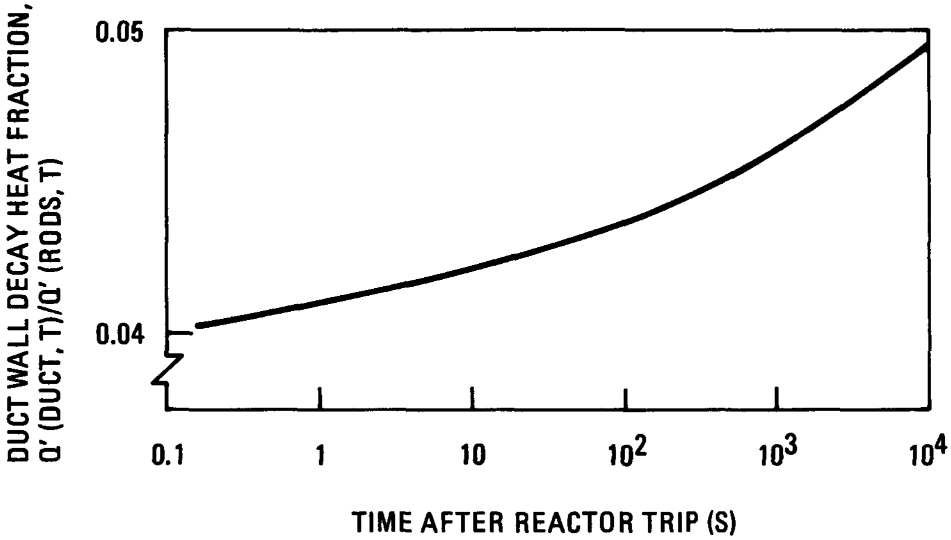

Fig. 5-9. Ratio of duct wall heating to fuel rod heating at time t

the rod release. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time that the rods are free and begin to fall. The time delay includes conservatively assumed values for relay response time of 0.10 s, the trip magnet release delay of 0.05 s, and sensing delays of 0.15 s for all trip parameters. For those parameter measurements with significant thermal inertia (i.e., steam and helium temperature measurements), a first-order lag simulates the sensor response. Table 5-9 gives the nominal trip and limiting trip setpoints assumed in the accident analyses, the time delays, and the sensor lags assumed for each trip function for the primary trip system. The difference between the limiting trip setpoint assumed for the analysis and the nominal trip setpoint represents an allowance for the instrumentation channel and the setpoint error. Nominal trip setpoints were used in the analysis of PC-1 and PC-2 events in conjunction with the best-estimate analysis model. Limiting trip setpoints were used for PC-3, PC-4, and PC-5 events for the conservative analyses.

The total negative reactivity worth available for reactor trip at the beginning-of-life (BOL) is $15.66 for the control rods. Additionally, for the PC-3 through PC-5 transient analyses discussed in this report, the most reactive rod with reactivity worth of $2.65 is assumed to be stuck at the fully withdrawn position when a primary reactor trip is actuated so that only $13.01 is available for reactor trip.

Secondary Reactor Trip System. The secondary trip system releases the shutdown rods using signals from the secondary trip parameters (Fig. 4-14). Section 4.4.2 details the secondary trip actuation.

Table 5-10 gives the nominal trip and the limiting trip setpoints assumed in the accident analyses and the time delays and sensor lags assumed for each trip function for the secondary trip system.

The total negative reactivity worth available for reactor trip at BOL is $13.60 for the shutdown rods. Additionally, for all PC-3 through PC-5

TABLE 5-9
PRIMARY TRIP PARAMETERS, LIMITING AND NOMINAL TRIP SETPOINTS, AND TRIP TIME DELAYS ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Setpoint	Nominal Trip Setpoint	Sensors Time Constant[a] (s)	Sensing Response Delay[b] (s)	Relay Response Delay[b] (s)	Trip Magnet Release Delay[b] (s)
Reactor power to feedwater flow ratio high	1.75	1.75	0.0	0.15	0.1	0.05
Reactor power high	114.5%	110%	0.0	0.15	0.1	0.05
Reactor power to core flow ratio high with the reactor at power	1.44	1.3	0.0	0.15	0.1	0.05
Primary coolant pressure low	8.80 MPa (1276 psia)	8.96 MPa (1300 psia)	0.0	0.15	0.1	0.05
Primary coolant moisture high	(later)	(later)	0.0	0.15	0.1	0.05
Containment pressure high	0.0407 MPa (20.9 psia)	0.0345 MPa (20 psia)	0.0	0.15	0.1	0.05
Manual reactor trip	--	--	--	--	0.1	0.05

[a] Time constant = exponential time constant for a first-order lag representation.

[b] Time delay = pure delay from receipt of signal to action.

TABLE 5-10
SECONDARY TRIP PARAMETERS, LIMITING AND NOMINAL TRIP SETPOINTS, AND TRIP TIME DELAYS ASSUMED IN ACCIDENT ANALYSES

Trip Function	Limiting Trip Setpoint	Nominal Trip Setpoint	Sensors Time Constant[a] (s)	Sensing Response Delay[b] (s)	Relay Response Delay[b] (s)	Trip Magnet Release Delay[b] (s)
Reactor containment isolation	--	--	0.0	0.15	0.1	0.05
Primary coolant pressure high	11.51 MPa (1669 psia)	11.34 MPa (1645 psia)	0.0	0.15	0.1	0.05
Reactor power high	119.5%	115%	0.0	0.15	0.1	0.05
Rate of reactor power increase high	32%/min	30%/min	5.0	0.15	0.1	0.05
Total feedwater flow low with the reactor at power	20%	10%	0.0	0.15	0.1	0.05
Loop A (or B or C) steam generator inlet temperature high	581°C (1077°F)	566°C (1050°F)	15.0	0.15	0.1	0.05
Loop A (or B or C) steam generator temperature rate of increase high	150°C/min (270°F/min)	139°C/min (250°F/min)	15.0	0.15	0.1	0.05
Manual reactor trip	--	--	--	--	0.1	0.05

(a)Time constant = exponential time constant for a first-order lag representation.

(b)Time delay = pure delay from receipt of signal to action.

analyses discussed in this report, the most reactive rod with reactivity worth of \$3.81 is assumed to be stuck at the fully withdrawn position when a secondary reactor trip is actuated so that only \$9.79 is available for reactor trip.

5.1.5.2. RHR Initiation System. As described in Section 4, the GCFR plant is equipped with a nonsafety RHR system, the MLCS, and two safety-related RHR systems, the SCS and CACS. The normal RHR system operation sequence is (1) MLCS, (2) SCS, and (3) CACS. However, in extremely unlikely events, such as DBDA, the SCS are not designed to be used.

The nonsafety MLCS can perform RHR for all anticipated and accident events. The SCS is a safety system and will be used for likely and unlikely events. The nonsafety MLCS and the CACS safety RHR systems can be applied for all likely, unlikely, and extremely unlikely events. Section 4.5 details RHR initiation system logic.

5.1.6. Computer Codes Utilized

Some of the principal computer programs used in transient analyses are summarized below. Appendix B describes a specialized program in which the modeling has been developed to simulate specific effects, such as the fuel assembly edge subchannels, with the respective analysis.

5.1.6.1. FASTRAN. The FASTRAN program incorporates models of the reactor core with kinetics, steam generators, main circulators, and the entire auxiliary cooling loop (including the auxiliary circulator and drive motor, core auxiliary heat exchanger (CAHE), auxiliary loop cooler (ALC), and all interconnecting piping). FASTRAN includes point model neutron kinetics and reactivity effects of the grid plate, fuel, and coolant. It represents the secondary side of the steam generator as a nodal boiler with regions of sub-cooled, saturated mixture, and superheated coolant for transient steam generator conditions. It simulates the reactor protection system to include reactor trips on neutron flux, overpower and reactor coolant overtemperature, high and low pressures, and low flow. It also simulates control systems (including rod control, steam generator isolation, and feedwater control). It models the safety actions (including reactor trip and safety RHR initiating systems under accident conditions).

For depressurization accident analyses, FASTRAN analyzes the coolant blowdown phase, utilizing an option* which incorporates detailed nonlinear models for the coolant dynamics to evaluate the incoming and outgoing reactor coolant flows associated with the containment atmosphere, the reactor inlet and outlet plenums, and the circulator plenums. The system of coupled, nonlinear reactor coolant pressure-density-flow equations and the circulator dynamical equation are solved implicitly, employing the generalized Newton method. Following the blowdown phase of a depressurization accident, the containment atmosphere can communicate through the leak passage, resulting in air ingress into the primary coolant. FASTRAN includes this air ingress due to natural convection, if any, and thermally-induced inhalation effects.

FASTRAN calculates axial and radial temperature distributions occurring in a transient for six representative fuel or blanket rods, which consist of an average-powered rod, a maximum-powered rod, and a maximum-powered rod with hot spot engineering factors for the fuel and the blankets. FASTRAN is a versatile program suited to accident evaluation, control studies, and parameter sizing (see Ref. 5-16).

5.1.6.2. RATSAM. The RATSUM program was developed to evaluate the transient thermal-hydraulic behavior of the commercial HTGR (Ref. 5-21). The analytical model of the program solves a set of ordinary differential

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*In the analyses herein, the GAFTRAN option was used. GAFTRAN assumes the uniform mass flow rate in the coolant circuit within the prestressed concrete reactor vessel (PCRV) at any given time and evaluates the leak flow based on the average coolant pressure in the PCRV and the average containment pressure. Errors due to this assumption are negligible for the rate of depressurization considered herein.

equations for conservation of mass, energy, and momentum which govern the entire flow system. The flow system is represented by subvolumes (nodes) having heat transfer components. The program includes models of the core with point kinetics, steam generator, main circulator, and interconnecting coolant channels. Since the RATSAM program can analyze unequal loops, it is primarily used in GCFR calculations for natural circulation core cooling analyses with various coolant loop conditions. Reference 5-21 gives a more detailed program description.

5.1.6.3. CNTB. The CNTB (Ref. 5-22) program calculates the containment temperature and pressure responses during a postulated depressurization accident. CNTB accounts for heat transfer to horizontal and vertical heat sink surfaces and effects of specified degrees of gas mixing between the containment air and the discharged helium. The CNTB results are used for the containment building design and the FASTRAN input for the coolant back pressure during a depressurization accident.

5.1.6.4. COBRA. The COBRA-IV (Ref. 5-23) program performs detailed thermal-hydraulic analyses of rod bundles. COBRA-IV has been used extensively for design analyses of GCFR fuel, blanket, and control assemblies. The code determines the enthalpy and flow distribution in the bundle for both steady-state and transient conditions. General Atomic (GA) modified the code to incorporate features required for GCFR analyses. Principal among these are helium gas properties, GCFR-specific thermal-hydraulic correlations, circumferential thermal conduction in fuel rod cladding, radiation heat transfer between rods and between rods and the duct surface, gamma heat generation in the duct wall, and rod cladding hot spot temperature evaluation routines.

5.2. DECREASE IN REACTOR COOLANT FLOW RATE

This section discusses a number of postulated faults which could result in a decrease in reactor primary coolant flow and presents detailed analyses for the most limiting cases.

The following events are discussed:

1. Partial loss of normal primary coolant flow due to a loop trip with failure of the plant to reduce load (Section 5.2.1.).

2. Complete loss of normal primary coolant flow due to loss of power to circulator motors (Section 5.2.2).

3. One circulator bearing seizure (Section 5.2.3).

Case 1 is ANS PC-2; Case 2, ANS PC-3; and Case 3, ANS PC-4 (see Section 5.1).

To ascertain how large a margin exists in the GCFR core cooling capability, analyses were performed for cases using available, redundant RHR systems that are not required to meet the PC-5 deterministic safety and reliability goals. Section 5.2.4 summarizes these results.

5.2.1. Partial Loss of Normal Primary Coolant Flow

5.2.1.1. Identification of Causes and Accident Description. A partial loss of normal primary coolant flow could result from a mechanical or electrical failure in one of the circulators and/or its motor or due to loss of feedwater in one loop. If the reactor is at power at the time of the accident, the partial loss of coolant circulation could rapidly increase the fuel cladding temperature and result in fuel cladding damage if the plant control system does not reduce the reactor power.

Normal circulator motor power is supplied through individual buses connected to the generator. When the generator trips, the buses are automatically transferred to the auxiliary transformer, which is supplied from offsite power lines. Following a reactor trip, the main circulators are ramped down by the controller to 30% of normal full power speed. This 30% speed is then maintained by the RHR control system.

The high power-to-flow trip, actuated by a two-out-of-three high power-to-flow signal, provides the necessary protection for a partial loss of primary coolant flow with a failure of the plant control to reduce load. A power-to-flow ratio (normalized to 1.0 at full-power steady state) of greater than 1.3 will trip the reactor. The RHR will be carried out by the remaining MLCS loops, as described in Section 4.5.1.

This event is ANS PC-2 (an incident of moderate frequency), as defined in Section 5.1.1.

5.2.1.2. Analysis of Effects and Consequences. The transient is analyzed using the detailed system computer program FASTRAN (Section 5.1.6.1). The following assumptions were made for this event:

1. RHR cooldown used two main loops.
2. Best-estimate model was applied (PC-2).
3. The tripped loop does not contribute to heat removal.

Section 5.1.3 describes plant characteristics and initial condition. The 100% steady-state full-power initial conditions were used for this analysis.

Due to the loop trip, the circulator in the disabled loop coasts down rapidly, thereby reducing the pressure rise in its compressor. This leads to a pressure reversal across the main loop isolation valve (MLIV), causing it to close. When the plant control system fails to reduce load, the reactor is tripped on the high power-to-flow signal, and the cooldown proceeds on the remaining MLCS.

Figure 5-10 shows the system response to a loop trip. Figure 5-10(a) illustrates the reactor power, coolant pressure, and core flow transients following loop trip with a failure of the plant control system to reduce load. The pressure decrease is due to core heat generation reduction following reactor trip (which occurs on high power-to-flow at ~2.9 s).

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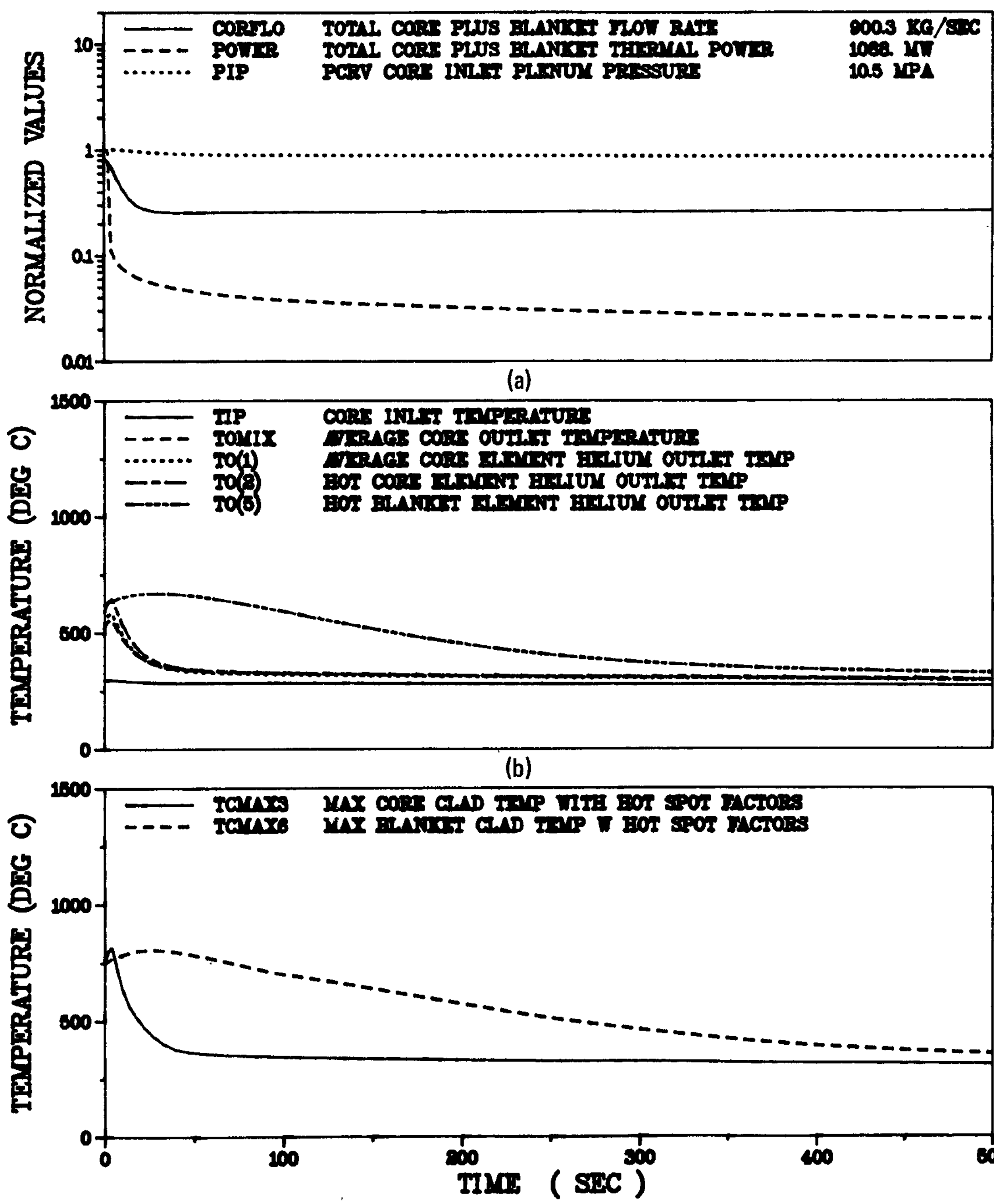

(c)

Fig. 5-10. Partial loss of primary coolant flow with two MLCS cooldown: (a) core power, flow, and pressure, (b) inlet and outlet gas temperature, (c) maximum cladding temperature (sheet 1 of 3)

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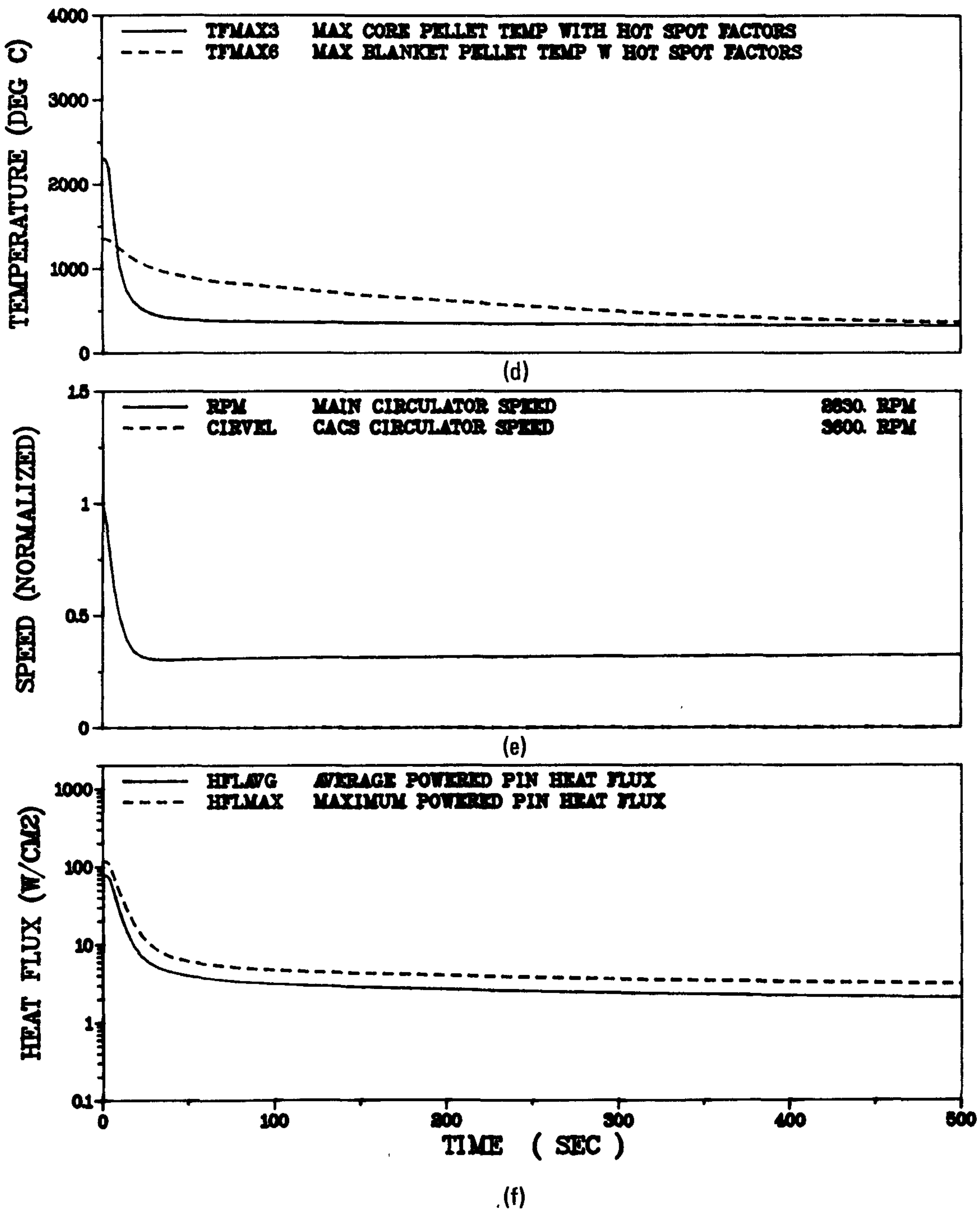

Fig. 5-10. Partial loss of primary coolant flow with two MLCS cooldown: (d) maximum fuel temperature, (e) circulator speeds, (f) cladding heat flux (sheet 2 of 3)

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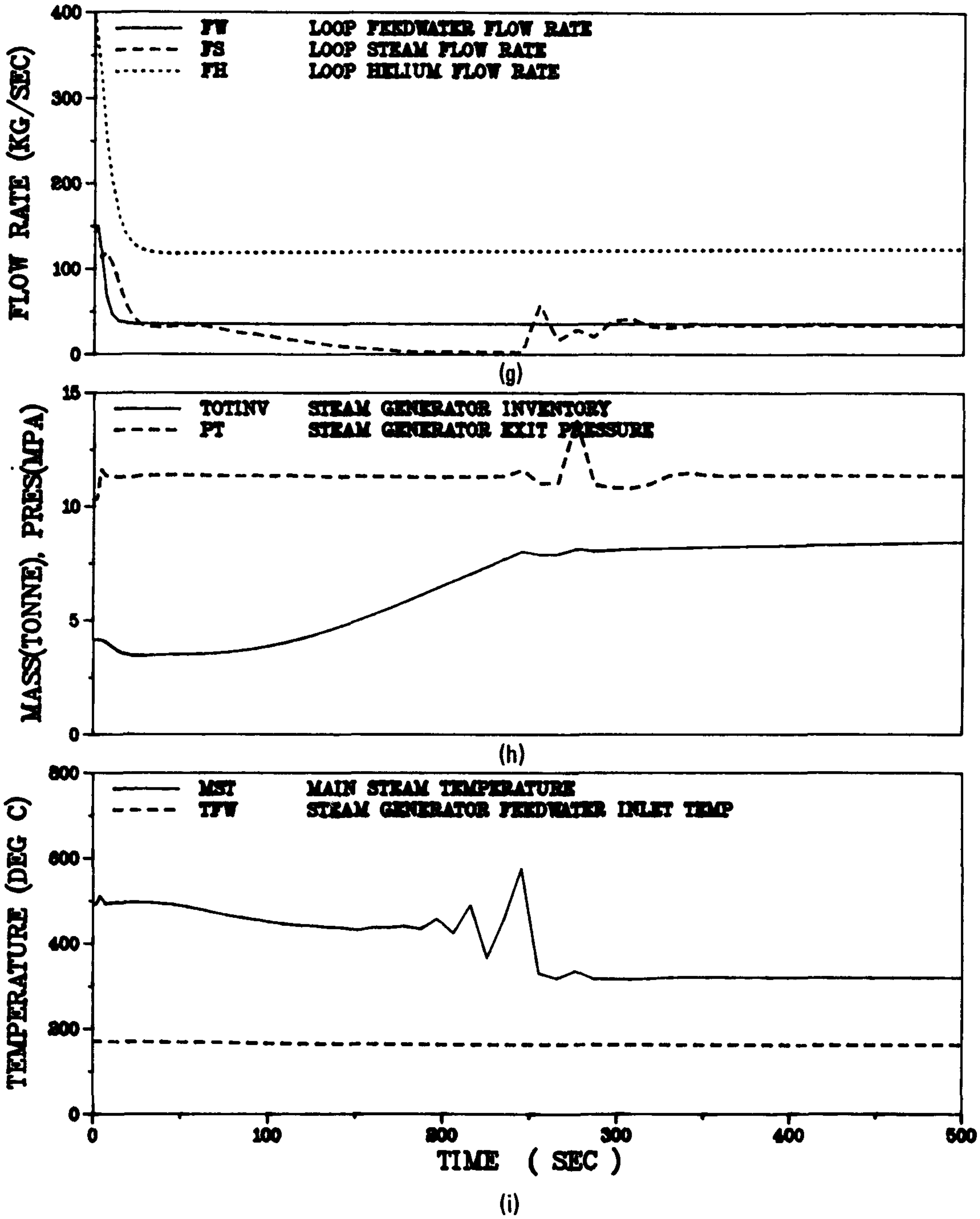

Fig. 5-10. Partial loss of primary coolant flow with two MLCS cooldown: (g) steam generator flow rates, (h) steam generator inventory and pressure, (i) steam generator temperatures (sheet 3 of 3)

The core flow rate decreases rapidly after reactor trip due to the coastdown of the main circulators. Figure 5-10(b) shows the primary coolant temperatures of the reactor inlet (TIP), the reactor mixed outlet (TOMIX), the average fuel channel outlet [TO(1)], the hot fuel channel outlet [TO(2)], and the hot blanket channel outlet [TO(5)]. The related component design limit, particularly the circulator, steam generator, and thermal barrier, as noted in Section 5.1.2, is adequately satisfied by the reactor mixed coolant outlet (TOMIX) and the cold leg (TIP) primary coolant temperatures. Figure 5-10(c) shows the transient response of the hot spot cladding temperatures of the maximum-powered fuel and blanket rods. Both rods reach maximum temperatures lower than the respective design temperature limits for PC-2 (see Section 5.1.2). Figure 5-10(d) shows maximum fuel temperature transient. Figure 5-10(e) shows the main circulator (RPM) rotative speed behavior during this transient. Figure 5-10(f) shows the transient core heat fluxes for the hot spot (HFLMAX) and the average (HFLAVG) fuel cladding surfaces.

Figure 5-10(g) summarizes the flow transients for the MLCS, particularly the feedwater flow rate (FW), primary coolant flow rate (FH), and steam flow rate (FS) in the steam generator. Figure 5-10(h) shows the steam generator steam/water inventory, and Fig. 5-10(i) shows the secondary coolant temperatures of the water (TFW) and steam (MST). The rapid change in main steam temperature at ~220 s indicates the floodout of the steam generator. Following floodout, the steam generator will operate in the single-phase mode, and the normal decay heat removal phase will be continued.

5.2.1.3. Radiological Consequences. The radiological consequences of this event are negligible, since primary coolant is not released to either the containment or the atmosphere.

5.2.1.4. Conclusions. The analysis results show that the high power-to-flow PPS signal and shutdown cooling by the remaining MCLS adequately protect against the partial loss of primary coolant flow event caused by single loop trip with a failure of the plant control system to decrease load. The maximum fuel and blanket rod cladding and the primary coolant temperatures are maintained below those PC-2 respective design limits. This event has negligible radiological consequences.

5.2.2. Complete Loss of Normal Reactor Primary Coolant Flow

5.2.2.1. Identification of Causes and Accident Description. A complete loss of forced normal reactor coolant flow could result from a simultaneous loss of all electrical power supplies to all main loop circulators. Normal power for the circulator motors is supplied through individual buses connected to the main turbine generator. When a main turbine generator trip occurs, the buses are automatically transferred to the auxiliary transformer, which is supplied from off-site power lines. If a consequent LOSP occurs, the main circulators will coast down, and the MLCS will be isolated. When the main circulator speed drops below 28% speed, or if the feedwater flow has not been reestablished within 20 s, the safety-grade SCS loops and equipment are activated, then the pony motors will then maintain circulator speed at 30% (see Section 4.5.2), while the SCS feedwater flow is maintained at 25%. This event is ANS PC-3 (an infrequent incident), as defined in Section 5.1.1.

The necessary protection from a total loss of forced primary coolant flow accident is provided by the PPS. Loss of circulator power is sensed by two out of three logic in each loop and initiates shutdown of each MLCS loop (Section 4.5.1). The reactor is subsequently tripped on either of two signals: high power to feedwater flow or low total feedwater flow (Sections 4.4.1 and 4.4.2) as a consequence of the MLCS shutdown.

5.2.2.2. Analysis of Effects and Consequences. The FASTRAN code was used to analyze this event (Section 5.1.6.1), calculating the core, primary coolant, and primary coolant component transient thermal-hydraulic behavior.

The following cases were analyzed:

1. Total loss of normal circulation followed by cooldown on three SCS loops.

2. Total loss of normal circulation followed by cooldown on two SCS loops.

The total loss of normal circulation with cooldown on two SCS loops was analyzed by assuming that during the coastdown phase all circulators are coasting down and that when the SCS resumes cooling only two SCS loops are available. Since this is a PC-3 event, the analysis was performed with the conservative assumptions, as described in Section 5.1.3.6.

Section 5.1.3 describes plant characteristics and initial conditions. The 102% steady-state full-power initial conditions were used for the analysis.

Figure 5-11 shows the transient response for the total LOFC with cooldown on three SCS loops. Figure 5-12 shows the transient response for the total LOFC with cooldown on two SCS loops. In both cases, the fuel clad temperature does not increase significantly above its initial temperature. Table 5-11 gives the calculated sequence of events for this accident. The main circulators will coast down until the PPS initiates start of the pony motors, then the SCS pony motors will maintain circulator speed at the predetermined level. With the reactor tripped, a stable plant condition will be attained. Normal plant shutdown may then proceed.

5.2.2.3. Radiological Consequences. Since no primary coolant is released to either the containment or the atmosphere, the radiological consequences of the total loss of primary coolant circulation are insignificant.

5.2.2.4. Conclusions. The analysis of this event shows that the core and blanket cladding and the primary coolant tempertures are maintained well below the limits for PC-3 (Table 5-4) during this event.

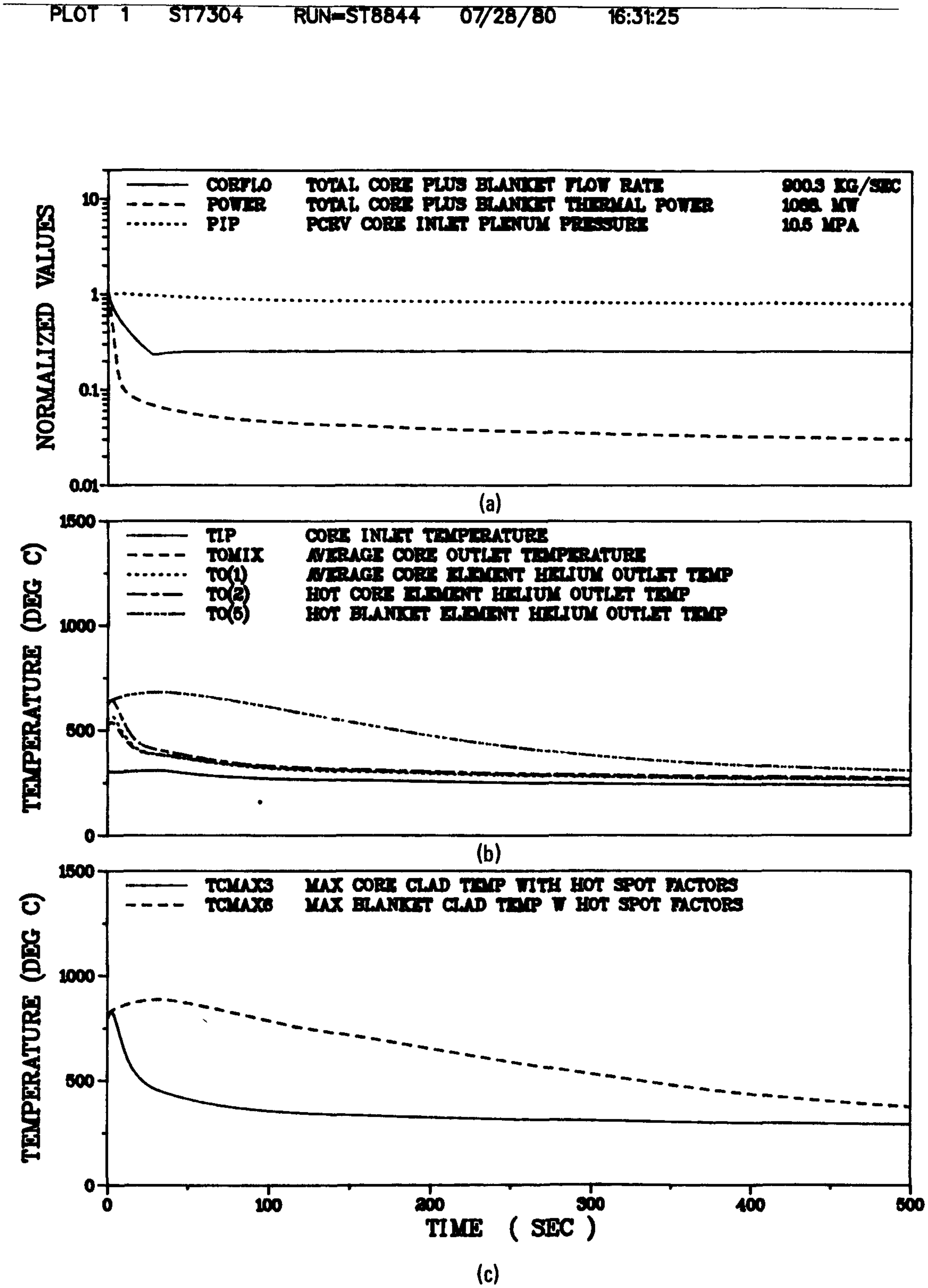

Fig. 5-11. Total loss of main circulator power with three SCS cooldown: (a) core power, flow, and pressure, (b) inlet and outlet gas temperatures, (c) maximum cladding temperatures

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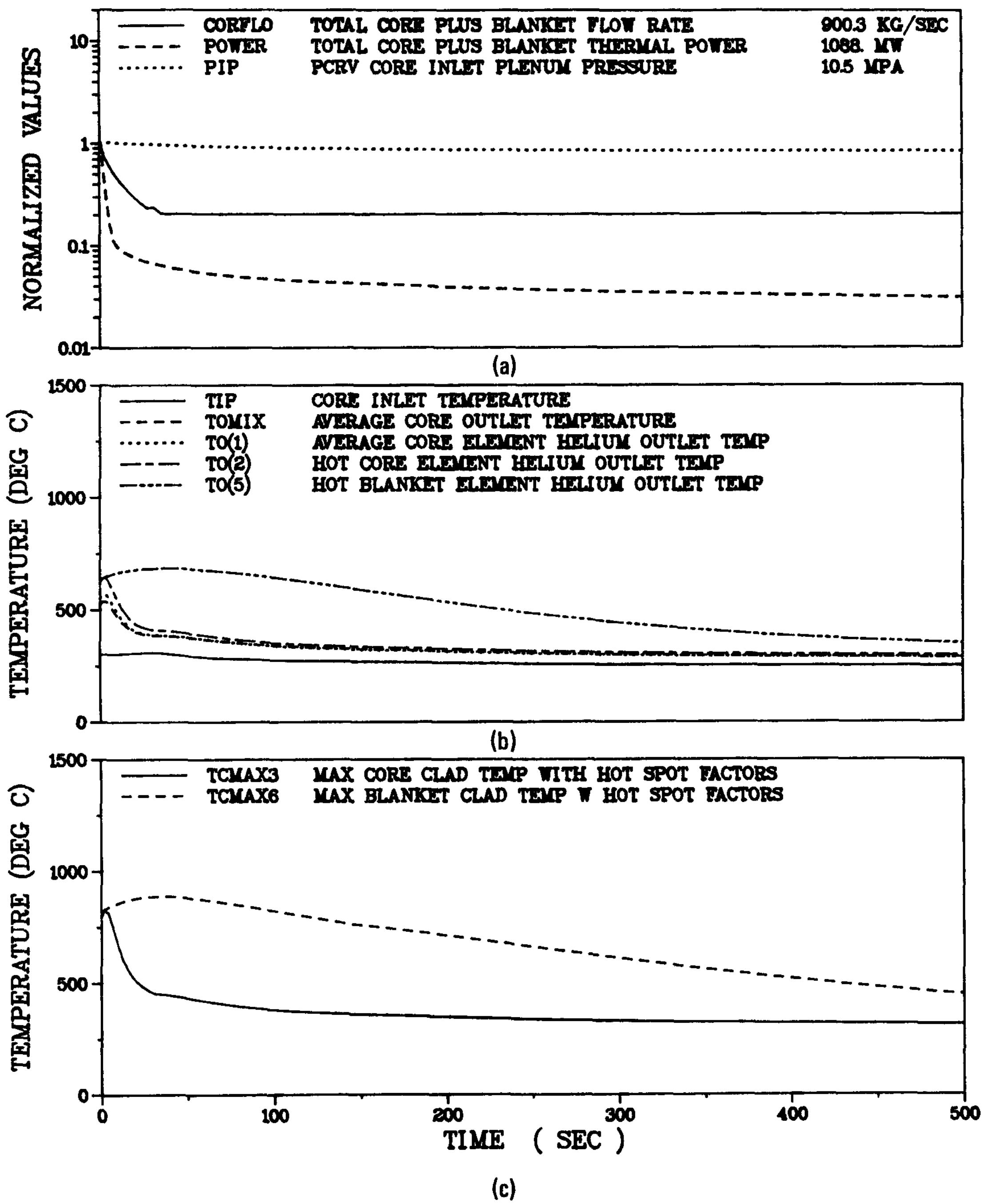

Fig. 5-12. Total loss of main circulator power with two SCS cooldown: (a) core power, flow, and pressure, (b) inlet and outlet gas temperatures, (c) maximum cladding temperatures

TABLE 5-11
CALCULATED SEQUENCE OF EVENTS FOR TOTAL LOSS OF MAIN CIRCULATOR POWER

Cooldown Time on Three SCS

(s)	Action
0	Loss of power to all main circulator motors
0 +	Loss of circulator power signal and MLCS isolation
0.4	Reactor trip
2.0	Peak fuel cladding temperature reached
20	Low feedwater flow over 20 s, signal obtained to initiate SCS
~100	SCS pony motors stablize both circulators at 30%

Cooldown Time on Two SCS

(s)	Action
0	Loss of power to all main circulator motors
0 +	Loss of circulator power signal and MLCS isolation
0.4	Reactor trip
2.0	Peak fuel clad temperature reached
20	Low feedwater flow over 20 s, signal obtained to initiate SCS
95	SCS pony motors stabilize circulator speed at 30%

5.2.3. Circulator-Bearing Seizure

5.2.3.1. Identification of Causes and Accident Description. The accident postulated is a seizure of a single main circulator. The primary coolant flow through the affected loop is postulated to stop almost instantly due to the low inertia of the primary coolant. The main loop primary coolant isolation valve would close due to the reverse pressure gradient imposed by the operating loops. Coupled with this event, power is assumed to be lost to all main circulators due to LOSP. The reactor is tripped as a consequence of MLCS loop trip due to loss of circulator power signal. Following the reactor trip, the two intact SCS loops would continue the core cooling. Following LOSP, the intact circulator would coast down, and the MLCS would be isolated, triggering the SCS initiation after ~20 s. The RHR would then proceed on the remaining two SCS.

If additional failure is assumed to occur in the remaining SCS, the RHR would then be performed by the remaining one SCS.

The above combination of events is ANS PC-5, as defined in Section 5.1.1.

5.2.3.2. Analysis of Effects and Consequences. This incident was analyzed using the FASTRAN code (Section 5.1.6.1), which calculated the core, primary coolant, and primary coolant component transient thermal-hydraulic behavior. The following cases were analyzed:

1. One circulator bearing seizure with cooldown on two SCS loops.

2. One circulator bearing seizure with cooldown on two SCS loops and a single failure (MLIV fails to close).

3. One circulator bearing seizure with cooldown on one SCS loop (assumes single failure is one of the remaining SCS loops).

power is also lost to the other circulators, and the MLCS is isolated. The initial cooling is assumed to be provided by the remaining circulators, which are assumed to be coasting down. Upon receiving the lack of feedwater flow for more than 20 s, the SCS is activated. Since this is an event beyond PC-3, the analysis was performed with conservative assumptions indicated in Section 5.1.3.6.

Section 5.1 describes plant characteristics and initial conditions. The 102% steady-state full-power conditions were used for the analysis.

Figure 5-13 shows the plant transient response to a circulator bearing seizure with cooldown on the remaining SCS loops. The reactor is tripped at 0.4 s following the initiating event, and the initial cooldown is on the intact loop due to coastdown of the circulator. At ~20 s, the SCS is initiated by the low feedwater signal for more than 20 s, and the cooldown proceeds on the remaining SCS loops.

Figure 5-14 shows the plant transient response to one circulator bearing seizure with cooldown on two SCS loops with an assumed single failure of a MLIV to close. This results in a large fraction of the primary coolant flow bypassing the core, which is evident when the core flows in Figs. 5-13(a) and 5-14(a) are compared. Figure 5-15 shows the plant transient response to one circulator bearing seizure with cooldown on one SCS loop. This assumes that the remaining one intact SCS loop takes over RHR duty. This event gives the highest cladding temperatures; however, the temperatures are still significantly lower than the allowable PC-5 limits (Table 5-12). Normal plant shutdown proceeds following the initiation of the SCS.

5.2.3.3. Radiological Consequences. Since no primary coolant is expected to be released to either the containment or the atmosphere during this event, no radiological consequences result.

5.2.3.4. Conclusions. The analyses show that none of the plant critical temperatures or pressures were reached during the transient. However, the

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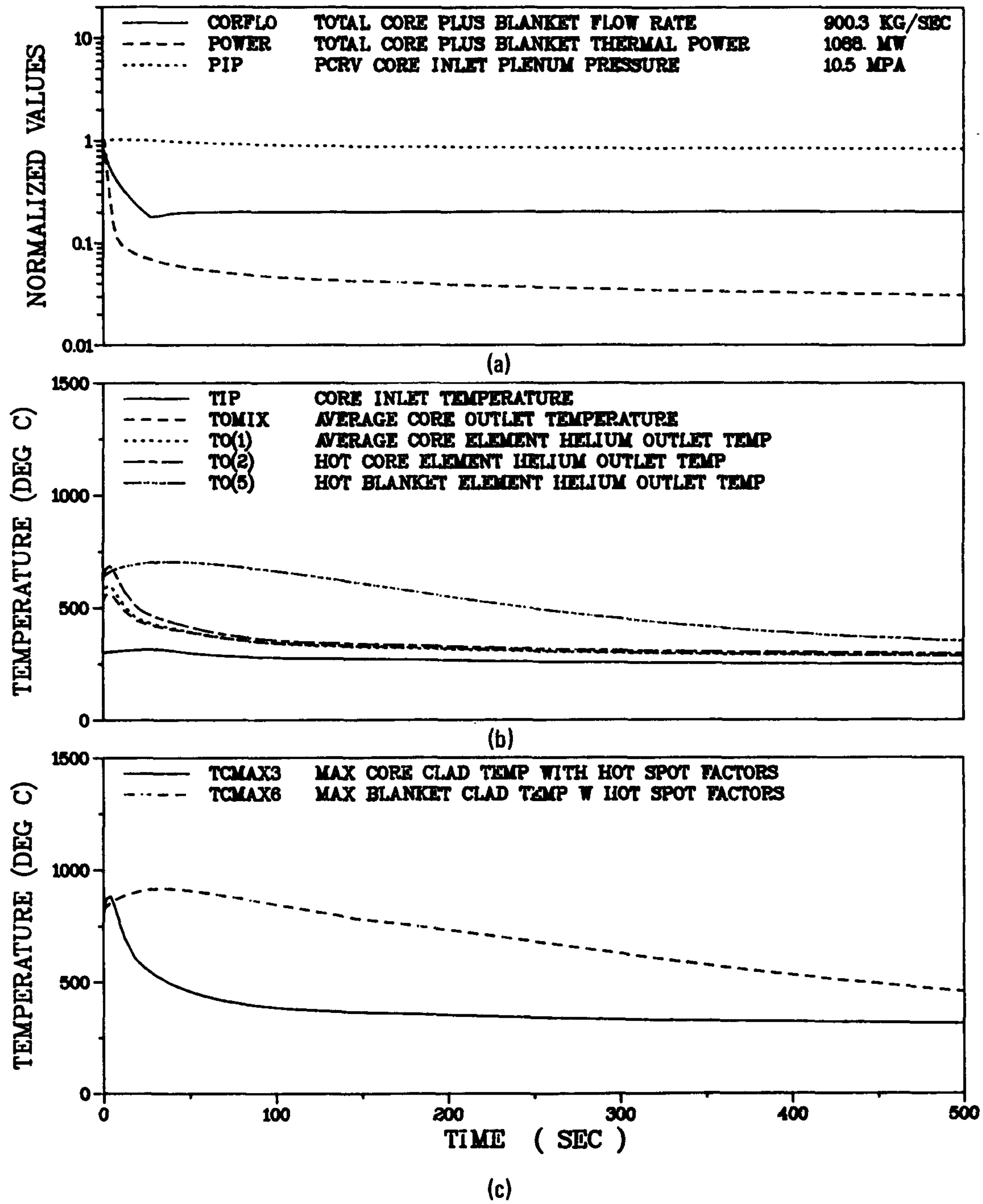

Fig. 5-13. Circulator bearing seizure with two SCS cooldown: (a) core power, flow, and pressure, (b) inlet and outlet gas temperatures, (c) maximum cladding temperatures

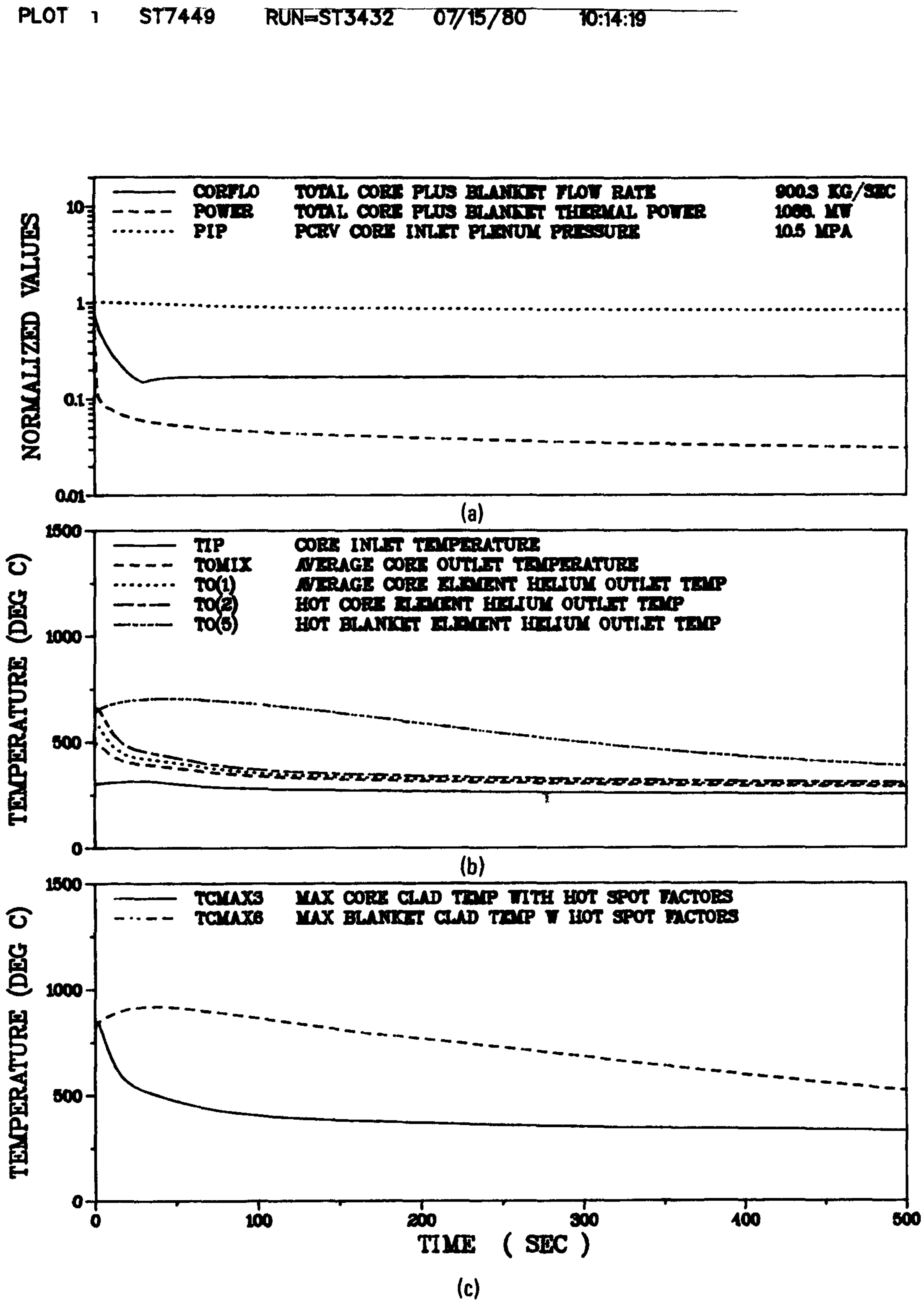

Fig. 5-14. Circulator bearing seizure with one MLIV stuck open, two SCS cooldown: (a) core power, flow, and pressure, (b) inlet and outlet gas temperatures (c) maximum cladding temperatures

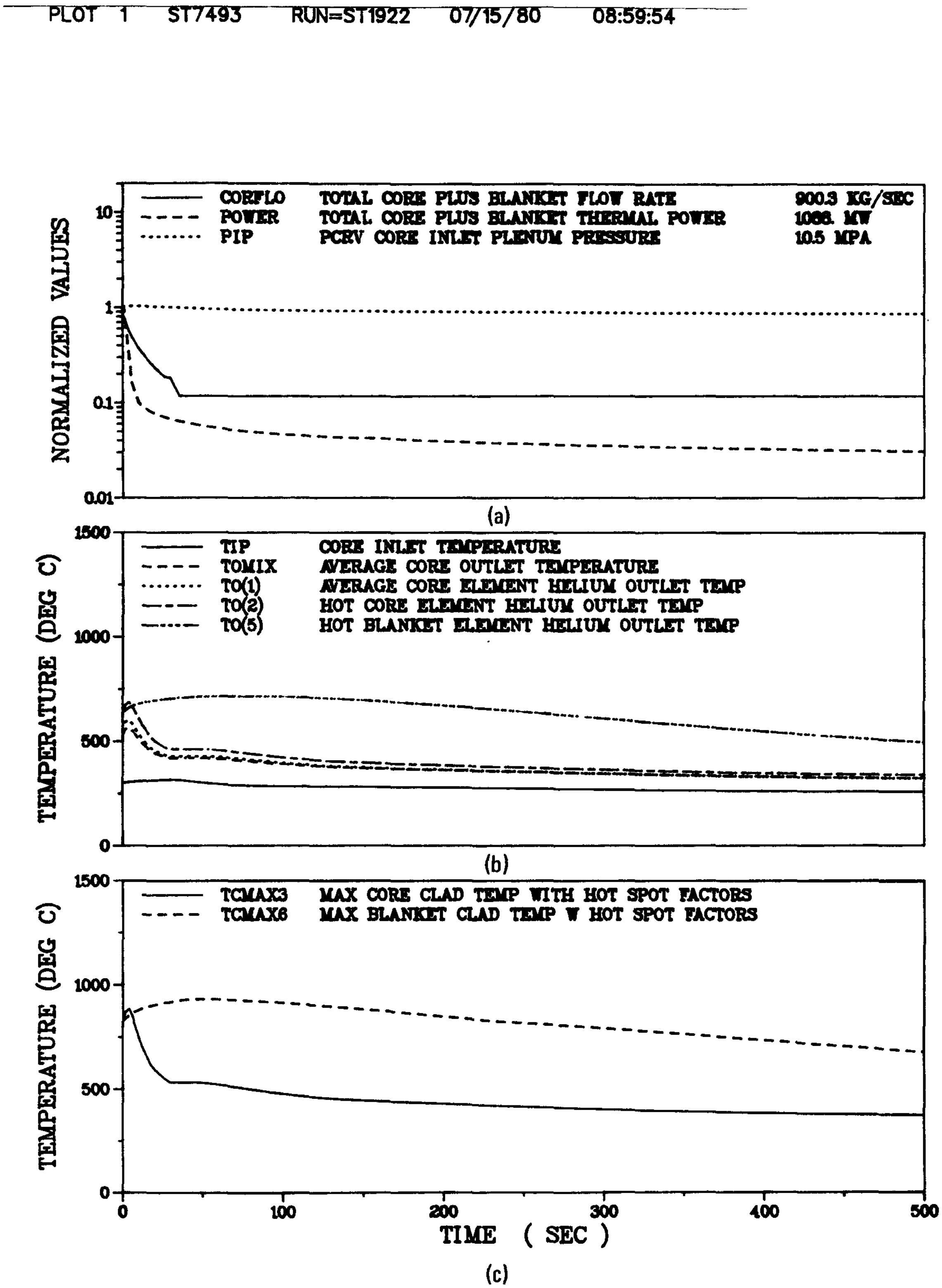

Fig. 5-15. Circulator bearing seizure with one SCS cooldown: (a) core power, flow, and pressure, (b) inlet and outlet gas temperatures, (c) maximum cladding temperatures

TABLE 5-12
CALCULATED SEQUENCE OF EVENTS FOR ONE CIRCULATOR BEARING SEIZURE

Cooldown Time on Two SCS Loops

(s)	Action
0	Circulator bearing seizure and MLCS isolation
0.4	Reactor trip on loss of circulator power
20	SCS initiated on low feedwater flow for over 20 s
100	SCS circulator speed stabilized

Cooldown Time on Two SCS Loops with MLIV Failure to Close

(s)	Action
0	Circulator bearing seizure, MLCS isolation, and MLIV stuck open
0.4	Reactor trip
20	SCS initiated on low feedwater flow low over 20 s
100	SCS circulator speed stabilizes on two SCS loops with one MLIV stuck open

Cooldown Time on One SCS

(s)	Action
0	Circulator bearing seizure and MLCS isolation
0.4	Reactor trip
3.0	Peak fuel clad temperature reached
20	SCS initiated on low feedwater flow over 20 s
50	Peak blanket cladding temperature reached
100	SCS circulator speed stabilized on remaining one SCS loop

cooldown on one SCS loop following one circulator bearing seizure is the limiting case for the decrease of primary coolant flow events. The maximum cladding hot spot temperatures reached were 890°C (1634°F) and 932°C (1707°F), respectively, for fuel and blanket pins, considerably below the allowable PC-5 limits. The primary coolant temperature transients indicate that the PC-5 limits for the essential loop components are also adequately met.

5.2.4. Additional RHR Capability Beyond Safety System Requirements

To indicate the large margin in RHR capabilities, this section summarizes the cases using redundant RHR systems that are available, but not required, for safety-related actions discussed in previous sections. Section 5.2.3 discussed bearing seizure cases; assuming LOSP and single failure, the cooldown was shown to be carried out by either two SCS loops with a MLIV failure to close as the single failure or one SCS loop, assuming the other remaining SCS loop failed as the single failure. As additional margin, the GCFR can tolerate an additional single failure such that a cooldown could be on either two CACS without an isolation valve failure or three CACS with one isolation valve failure.

In both cases, the peak cladding temperatures are well below the allowable limit.

5.2.5. Summary and Conclusion for Category of Decrease in Reactor Coolant Flow

To provide a perspective for depth of protection provided by the RHR systems, Figs. 5-16 and 5-17 summarize the results of all cases of this event catagory, including the margin cases. Figures 5-16 and 5-17 show the maximum fuel and blanket cladding temperatures, respectively, with an abscissa indicating number and type of the RHR system loops used. Dark symbols signify the margin cases which assume multiple failures beyond the deterministic safety requirements.

SYMBOL	EVENT	ADDITIONAL FAILURES	MAX CLADDING TEMPERATURE (^oC)	LIMIT TEMPERATURE (^oC)	PLANT CONDITION
△	LOOP TRIP WITHOUT POWER RUNBACK	1	826	850	PC-2
□	LOSS OF ALL CIRCULATOR POWER, LOSP	0	758	950	PC-3
◇	LOSS OF ALL CIRCULATOR POWER, LOSP	1	832	1100	PC-4
▽	CIRCULATOR BEARING SEIZURE, LOSP	0	890	1100	PC-4
○	CIRCULATOR BEARING SEIZURE, LOSP	1	856	1300	PC-5
●	CIRCULATOR BEARING SEIZURE, LOSP	1	890	1300	PC-5
▲	CIRCULATOR BEARING SEIZURE, LOSP	2	888	1300	BEYOND PC-5
■	CIRCULATOR BEARING SEIZURE, LOSP	3	888	1300	BEYOND PC-5
◆	CIRCULATOR BEARING SEIZURE, LOSP	3	888	1300	BEYOND PC-5

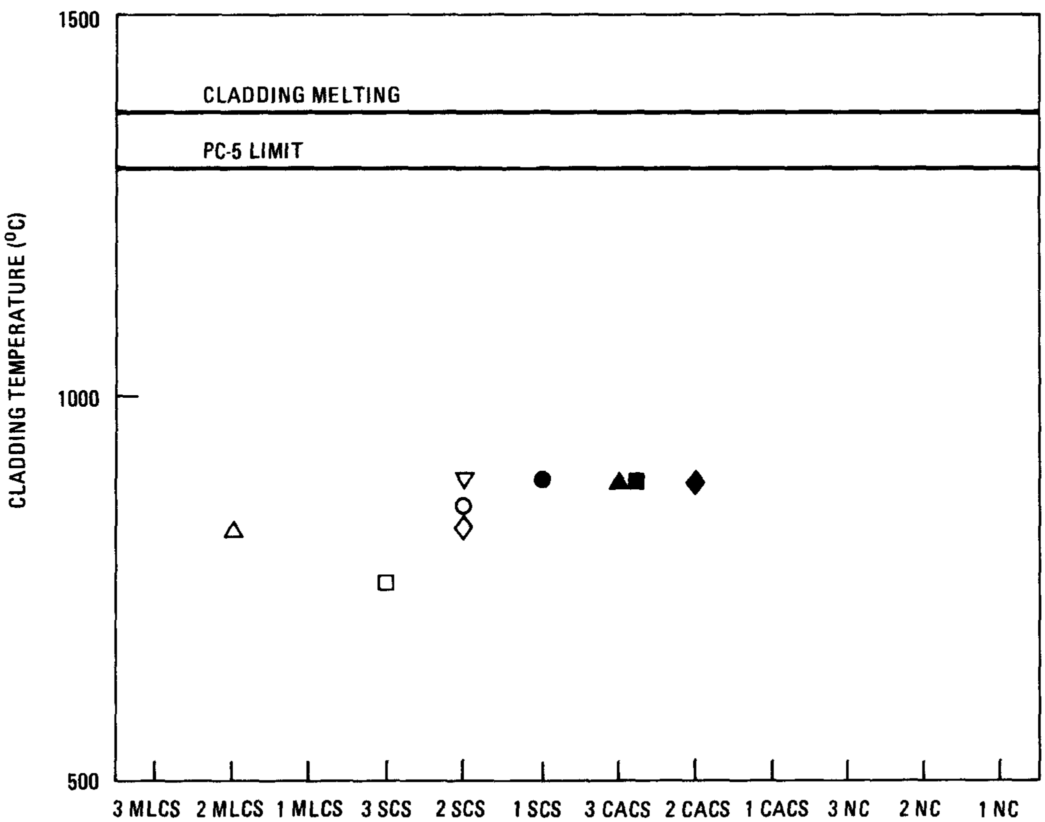

Fig. 5-16. Summary of core cooling performance in event category of decrease in reactor coolant flow rate; maximum fuel cladding temperature

SYMBOL	EVENT	ADDITIONAL FAILURES	MAX CLADDING TEMPERATURE (^oC)	LIMIT TEMPERATURE (^oC)	PLANT CONDITION
△	LOOP TRIP WITHOUT POWER RUNBACK	1	804	850	PC-2
□	LOSS OF ALL CIRCULATOR POWER, LOSP	0	791	950	PC-3
◇	LOSS OF ALL CIRCULATOR POWER, LOSP	1	888	1100	PC-4
▽	CIRCULATOR BEARING SEIZURE, LOSP	0	918	1100	PC-4
○	CIRCULATOR BEARING SEIZURE, LOSP	1	920	1300	PC-5
⊙	CIRCULATOR BEARING SEIZURE, LOSP	1	932	1300	PC-5
▲	CIRCULATOR BEARING SEIZURE, LOSP	2	933	1300	BEYOND PC-5
■	CIRCULATOR BEARING SEIZURE, LOSP	3	934	1300	BEYOND PC-5
◆	CIRCULATOR BEARING SEIZURE, LOSP	3	936	1300	BEYOND PC-5

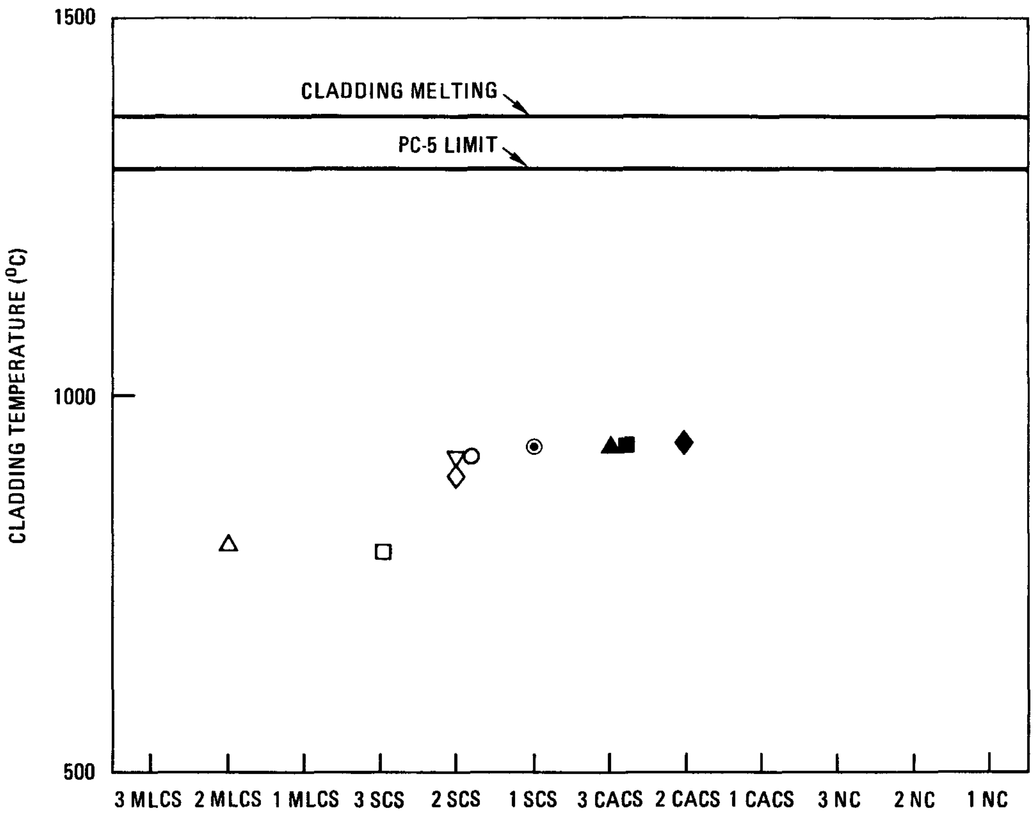

Fig. 5-17. Summary of core cooling performance in event category of decrease in reactor coolant flow rate; maximum blanket cladding temperature

These figures indicate adequate core cooling performance and significant depth of redundancy in available backup RHR systems.

5.3. DECREASE IN REACTOR HEAT REMOVAL BY THE SECONDARY SYSTEM

This section discusses events which could reduce the secondary system capacity to remove heat generated in the reactor coolant system. This section presents the results of detailed analyses for the limiting cases in this category.

In general, the initiating event leads to reactor cooldown by the MLCS. Coincident occurrences and/or single failures coupled with the initiating event can lead to a reduction or loss of the MLCS availability. This requires using the SCS and CACS backup cooling systems (detailed in Section 4.3) for reactor cooldown. This section presents analyses for the initiating events and the progressive failure cases:

1. Loop trip (Section 5.3.1).
2. Turbine trip (Section 5.3.2).
3. Loss of condenser vacuum (Section 5.3.3).
4. Inadvertant steam generator isolation (Section 5.3.4).
5. LOSP (Section 5.3.5).
6. Loss of normal feedwater flow (Section 5.3.6).

To show the large margin in core cooling capability, additional analyses were performed for cases using redundant RHR systems. These systems are available, but not required, to meet the PC-5 safety and reliability goal. Section 5.3.7 summarizes these results.

5.3.1. Loop Trip

5.3.1.1. Identification of Causes and Accident Description. A loop trip results in a partial loss of main loop cooling capability. The possible reasons for a loop trip are discussed in Section 4.5.1.3. Reactor and turbine trips are initiated by high power-to-flow or high power-to-feedwater signal. Main feedwater flow of the operative loops is decreased to ~25% of full-load loop feedwater flow, and reactor cooldown occurs on the two remaining main loops. This event with MLCS loop cooldown is ANS PC-2.

If a coincident LOSP occurs during loop trip, cooldown in this case would use the SCS with feedwater flow through the two operating loops. With a cooldown on two SCS loops, the accident is ANS PC-3.

When a single failure in the SCS loop is postulated after the loop trip and coincident LOSP, the cooldown must be performed with a single SCS loop. This event is ANS PC-4.

5.3.1.2. Analysis of Effects and Consequences. The plant transient which occurs after a loop trip is similar to that following the partial loss of primary coolant event. Refer to Section 5.2.1.2 for the analysis of effects, results, and conclusions.

5.3.2. Turbine Trip

5.3.2.1. Identification of Causes and Accident Description. A turbine trip results in a rapid closure of the main turbine stop valve, stopping the steam flow to the turbine. Possible causes for the initiation of a turbine trip include the following:

1. Loss of condenser vacuum.
2. Generator trip.
3. Low bearing oil.
4. Turbine thrust bearing failure.
5. Turbine overspeed.
6. Low main steam pressure or temperature.
7. Manual trip.

Upon closing the turbine stop valve, feedwater flow is programmed to decrease rapidly to ~25% of full-load flow and the three main MLCS loops continue to cool the reactor.

After the stop valve is closed, the steam flow from the steam generator is bypassed around the main turbine, through a desuperheater and a flash tank, and into the condenser. When the condenser is out of service, the steam flow is discharged to the atmosphere through the relief valve.

If a coincident LOSP should occur in addition to the turbine trip, a reactor trip occurs and the cooldown is performed with the SCS using all three cooling loops. This event is ANS PC-3.

A single failure in the SCS results in the loss of one of the cooling loops, and the cooldown must be performed on two SCS loops. This situation is ANS PC-4.

5.3.2.2. Analysis of Effects and Consequences. The plant behavior after a turbine trip with coincident LOSP is the same as with a normal reactor trip, since a turbine trip and reactor trip occur together in both cases.

5.3.3. Loss of Condenser Vacuum

5.3.3.1. Identification of Causes and Accident Description. Loss of condenser vacuum can cause a turbine trip. Section 5.3.2 describes turbine trip-initiating events. A loss of condenser vacuum results in the loss of the normal heat sink for the MLCS and a loss of condensate flow; therefore, the resulting event is the same as a turbine trip with LOSP described in Section 5.3.2.1. Cooldown for this event is normally performed on the SCS. This case is ANS PC-2.

A single failure in the SCS results in the loss of one SCS cooling loop, so two SCS loops perform cooldown. This event is ANS PC-3.

In case transfer to the SCS system fails because of a PPS failure or operator action, the cooldown occurs on the CACS system. This cooldown is performed with three CACS loops and is ANS PC-3. If a single failure of one CACS loop occurs, cooldown is performed on the remaining two CACS loops and is ANS PC-4.

Possible causes of a vacuum loss in the condenser are air leakage, ejector malfunction, failure of the condenser cooling water pumps, or failure of the condensate pumps.

5.3.3.2. Analysis of Effects and Consequences. The plant behavior after a loss of condenser vacuum is basically the same as the loss of normal feedwater event. Section 5.3.6 presents analysis for this event.

5.3.4. Inadvertent Steam Generator Isolation

5.3.4.1. Identification of Causes and Accident Description. An inadvertent steam generator isolation would occur if the steam generator isolation valves close during normal plant operation. This could result from a valve failure, a control system malfunction, or a spurious electrical signal to the valve controller. The isolation valves in one of the main cooling loops are assumed to fail in the closed position. This is, in effect, a loss of a single steam generator. The reduced loop feedwater flow would be detected by the PPS, which would initiate a loop trip. If the reactor is subsequently tripped, the cooldown after the trip proceeds on the remaining main loops, similar to the sequence described for the loop trip in Section 5.3.1. This event is ANS PC-2.

In the event of a coincident LOSP occurring at the time of the loop trip cooldown, two SCS loops will perform RHR using the two remaining main steam generators. The inadvertent steam generator isolation with a coincident LOSP is ANS PC-3. The plant transient after this event is similar to the single loop with failure a coincident LOSP and cooldown on two SCS presented in Section 5.2.

If a single failure in one of the remaining SCS loops is postulated along with the loop isolation and the coincident LOSP, cooldown occurs on the remaining SCS loop. This event with the cooldown on one SCS loop is ANS PC-4.

5.3.4.2. Analysis of Effects and Consequences. The plant transient behavior after an inadvertent steam generator isolation is essentially the same as the loop trip transient. Refer to Section 5.2.1.2 for analysis of the effects and results of a loop trip with cooldown on two main loops.

5.3.5. LOSP

5.3.5.1. Identification of Causes and Accident Description. A complete loss of off-site ac power could result in the loss of all power to the main circulators and the plant auxiliaries, such as condensate pumps. This loss of power may be caused by a loss of the off-site grid or by a failure of the on-site ac power distribution system.

With the LOSP event, the power system protection relay system will automatically transfer the circulators and the auxiliaries to house power. The plant will then continue to operate on the MLCS.

In the event of a coincident turbine trip with the LOSP, both the house power and off-site power are lost. In this case, all power to the main circulators is lost, and cooldown is performed on three SCS loops, powered by the emergency diesel generators. This event is ANS PC-3.

For the case of a single failure in addition to the LOSP and turbine trip, one of the SCS loops is postulated to fail. In this case, which is ANS PC-4, the cooldown occurs on the two remaining SCS loops.

5.3.5.2. Analysis of Effects and Consequences. For the LOSP, complete shutdown of the reactor is not required. Since the turbine remains on-line, house power is available to operate the main circulators and other auxiliaries. Although a reactor power reduction to ~25% is required, the plant operation continues on the main loops. This transient is essentially the same as a normal plant load change transient and was not analyzed as a case involving a reduction of the capacity of the secondary system to remove heat.

In the case of LOSP and turbine trip and loss of one SCS loop as single failure, cooldown is on the two remaining SCS loops. The transient plant behavior is similar to the loss of normal feedwater flow and cooldown on two SCS loops discussed below. Refer to Section 5.3.6.2 for the analysis of effects, results, and conclusions.

5.3.6. Loss of Normal Feedwater Flow

5.3.6.1. Identification of Causes and Accident Description. A loss of normal feedwater flow can occur as a result of feedpump failures, loss of off-site ac power, feedwater valve malfunctions, or feedwater line breaks or leaks. The inadvertent steam generator isolation described in Section 5.3.4 is also a loss of feedwater flow event.

A partial loss of feedwater flow (due to a feedwater leak or a valve or pump malfunction) is detected by the PPS when the power-to-feedwater flow ratio reaches the PPS setpoint. An immediate trip is initiated, and cooldown is performed on the main cooling loops. Feedwater flow is reduced to ~25% of full-load flow for this cooldown.

If the malfunction prevents operation on the main loops or if an LOSP occurs coincident with the partial loss of feedwater, the cooldown is performed on the SCS system. An additional single failure results in the loss of one SCS loop, and the cooldown is performed on the two remaining SCS loops. Feedwater flow is maintained at ~25% of full load flow for the SCS operation.

The worst postulated loss of normal feedwater flow event is a complete loss of feedwater flow. Possible causes of this event are complete failure of the main boiler feedpump or a spurious closure of the main feedwater valves. A total loss of feedwater with a coincident LOSP is ANS PC-4; with a single failure (one SCS loop fails), it is ANS PC-5. In a total loss of feedwater, where the initiating event prevents the use of the MLCS, the core is cooled down on three SCS loops. This event is ANS PC-3. With a single failure, cooldown is performed by two SCS loops, and this case is ANS PC-4.

The analysis for the case of cooldown on two SCS loops is discussed below.

5.3.6.2. Analysis of Effects and Consequences. An analysis using the FASTRAN code, as described in Section 5.1.6.1, was performed to obtain the plant transients following a loss of feedwater flow. Plant behavior was evaluated for cooldown on two SCS loops. The major assumptions used in these analyses were as follows:

1. The plant was operating at 102% power, with feedwater flow and steam temperature with initial conditions, as described in Section 5.1.3.2.

2. Conservative factors were used on pressure drops, heat transfer coefficients, and other parameters, as described in Section 5.1.3.6.

3. Loss of feedwater flow is assumed to occur instantaneously, causing an immediate loss of the main loops.

4. The reactor trip and turbine trip are assumed to be initiated by a PPS signal after a detection of high core power-to-feedwater flow ratio.

5. The SCS system is assumed to take 28 s to come on-line, and the SCS feedwater supply increases from zero to full flow (25% of full load feedwater flow) in 2 s.

Figure 5-18 presents plant transients for the loss of normal feedwater flow with coincident LOSP followed by a cooldown on two SCS loops.

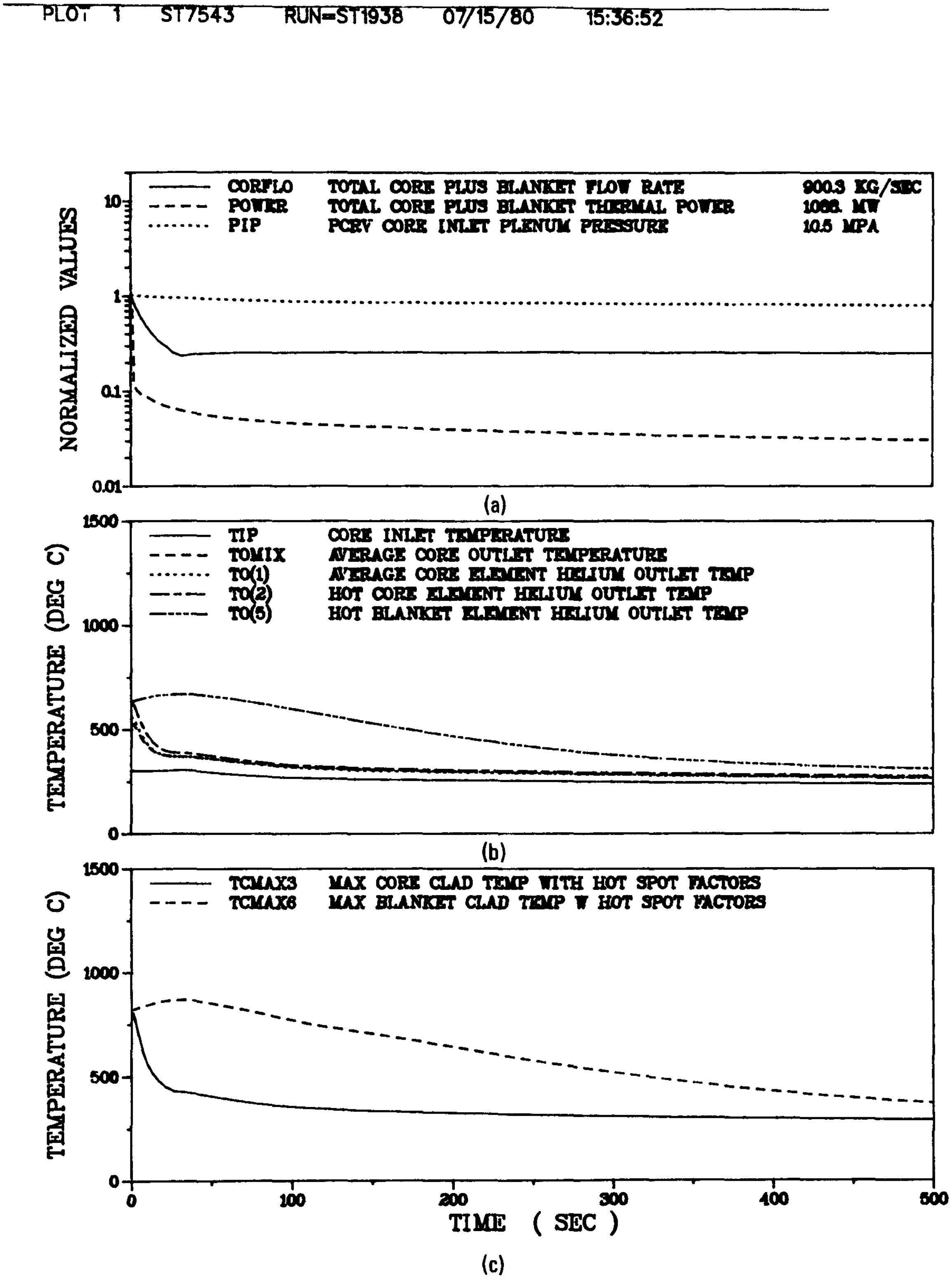

Fig. 5-18. Total loss of feedwater with two SCS cooldown: (a) core power, flow, and pressure, (b) inlet and outlet gas temperatures, (c) maximum cladding temperatures

The maximum fuel and blanket rod cladding temperatures in Fig. 5-18(c) indicate adequate margin relative to their PC-4 design limits in Table 5-4. The reactor mixed outlet (TOMIX) and the cold leg (TIP) primary coolant temperatures shown in Fig. 5-18(b) are low enough to prevent the essential components, circulators, steam generators, and thermal barriers from being exposed to their design limit temperatures specified in Table 5-4.

5.3.6.3. Radiological Consequences. For this event, no primary or secondary fluid is released. Therefore, no radiological consequences result from this accident.

5.3.6.4. Conclusions. Analysis results show that the loss of normal feedwater flow is not significant in terms of high core or helium temperatures. For the case studied, the maximum temperatures were lower than the corresponding full-load operating temperature. Also, no radiological consequences resulted from this event.

5.3.7. Additional RHR Capability Beyond Safety System Requirement

To indicate the large margin in the RHR capability, this section summarizes cases using redundant RHR systems. These redundant systems are available, but not required, for the safety-related actions discussed in previous sections.

The basic case, loss of feedwater flow (same transient as loss of condenser vacuum), involves a reduction in the secondary system capacity to remove heat. For this case, the following additional failures were postulated to arrive at cases which demonstrate the margin of RHR capability:

1. The transfer to SCS fails to occur; a failure is assumed to occur, resulting in the loss of one CACS loop; and one MLIV is assumed to remain stuck open.

2. The transfer to SCS fails to occur, and two CACS loops are assumed to fail.

In both the cases, the failures are assumed to occur coincident with the initiating event; therefore, all normal and SCS feedwater flows are assumed to be zero.

The plant parameter transients for the first margin case (i.e., the loss of condenser vacuum with cooldown on two CACS loops) are similar to the case described in Section 5.3.3, except that this case cools down on two CACS loops and some helium flow bypasses the core through the open loop.

The plant parameter transients for the second margin case (i.e., the loss of condenser vacuum with cooldown on one CACS loop) is similar to the case described above.

For both margin cases, the transient fuel, the blanket cladding, and the primary coolant temperatures are low enough to meet even the PC-1 design limits for the core and for the essential components in Table 5-4, even though the probabilities of this marginal sequence of events should be lower than those for PC-5.

5.3.8. Summary and Conclusion for Category of Decrease in RHR by Secondary System

To provide a perspective for depth of protection provided by the RHR systems, Figs. 5-19 and 5-20 summarize the results of all cases of this event category, including the margin cases. Figures 5-19 and 5-20 show the maximum fuel and blanket cladding temperatures, respectively, with an abscissa indicating number and type of the RHR system loops used. Dark symbols signify the margin cases which assume multiple failures beyond the deterministic safety requirements.

These figures indicate adequate core cooling performance and significant depth of redundancy in available backup RHR systems.

SYMBOL	INITIATING EVENT	ADDITIONAL FAILURES	MAX CLADDING TEMPERATURE (°C)	LIMIT TEMP (°C)	PLANT CONDITION
◻	LOSS OF ALL FEEDWATER	1	807	1100	PC-4
▼	LOSS OF CONDENSER VACUUM	3	807	1300	BEYOND PC-5
●	LOSS OF CONDENSER VACUUM	4	807	1300	BEYOND PC-5
▲	LOSS OF CONDENSER VACUUM	4	807	1300	BEYOND PC-5
■	LOSS OF CONDENSER VACUUM	5	807	1300	BEYOND PC-5
◆	LOSS OF CONDENSER VACUUM	5	807	1300	BEYOND PC-5

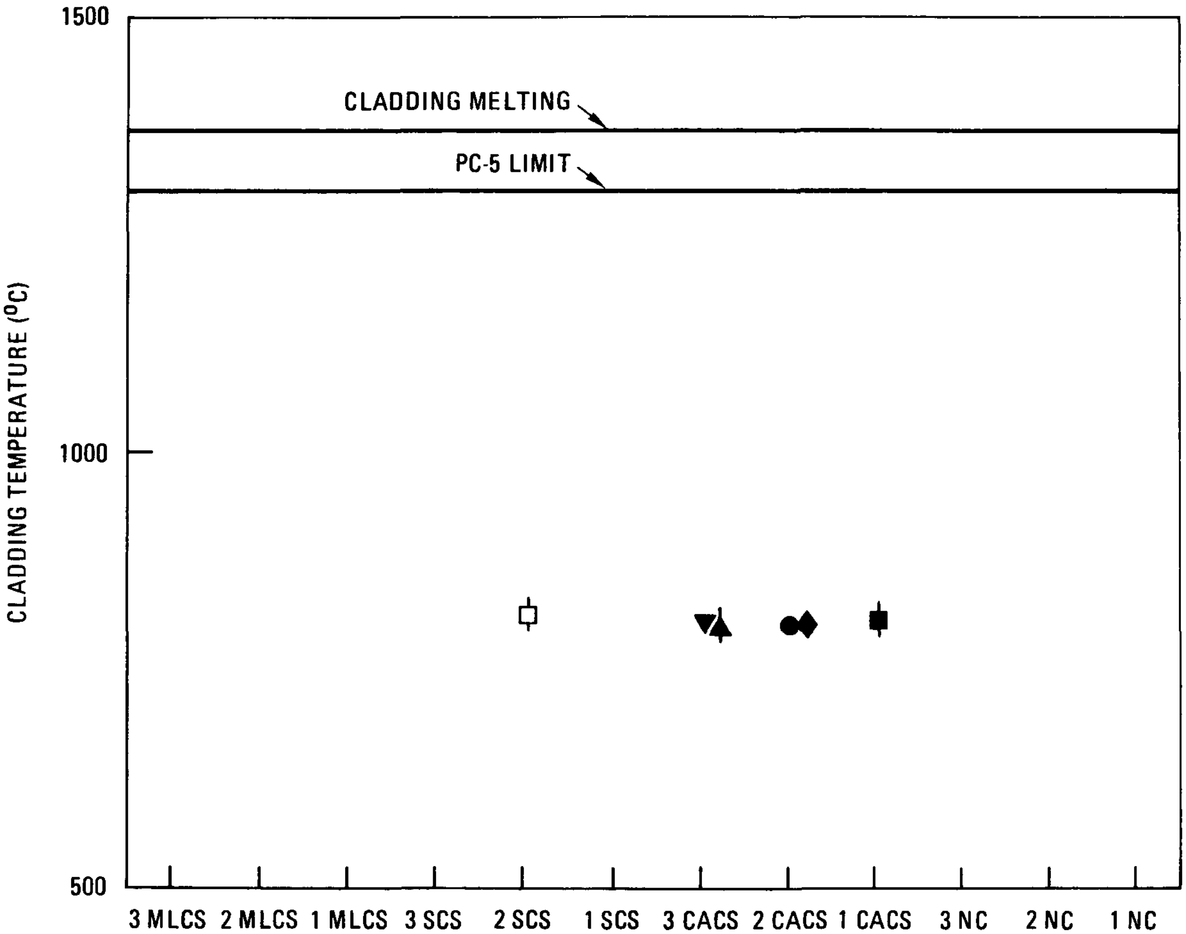

Fig. 5-19. Summary of core cooling performance in event category of decrease in reactor heat removal by secondary system; maximum fuel cladding temperature

SYMBOL	INITIATING EVENT	ADDITIONAL FAILURES	MAX CLADDING TEMPERATURE (^{o}C)	LIMIT TEMP (^{o}C)	PLANT CONDITION
◻	LOSS OF ALL FEEDWATER	1	871	1100	PC-4
▼	LOSS OF CONDENSER VACUUM	3	876	1300	BEYOND PC-5
●	LOSS OF CONDENSER VACUUM	4	878	1300	BEYOND PC-5
▲	LOSS OF CONDENSER VACUUM	4	878	1300	BEYOND PC-5
■	LOSS OF CONDENSER VACUUM	5	894	1300	BEYOND PC-5
◆	LOSS OF CONDENSER VACUUM	5	882	1300	BEYOND PC-5

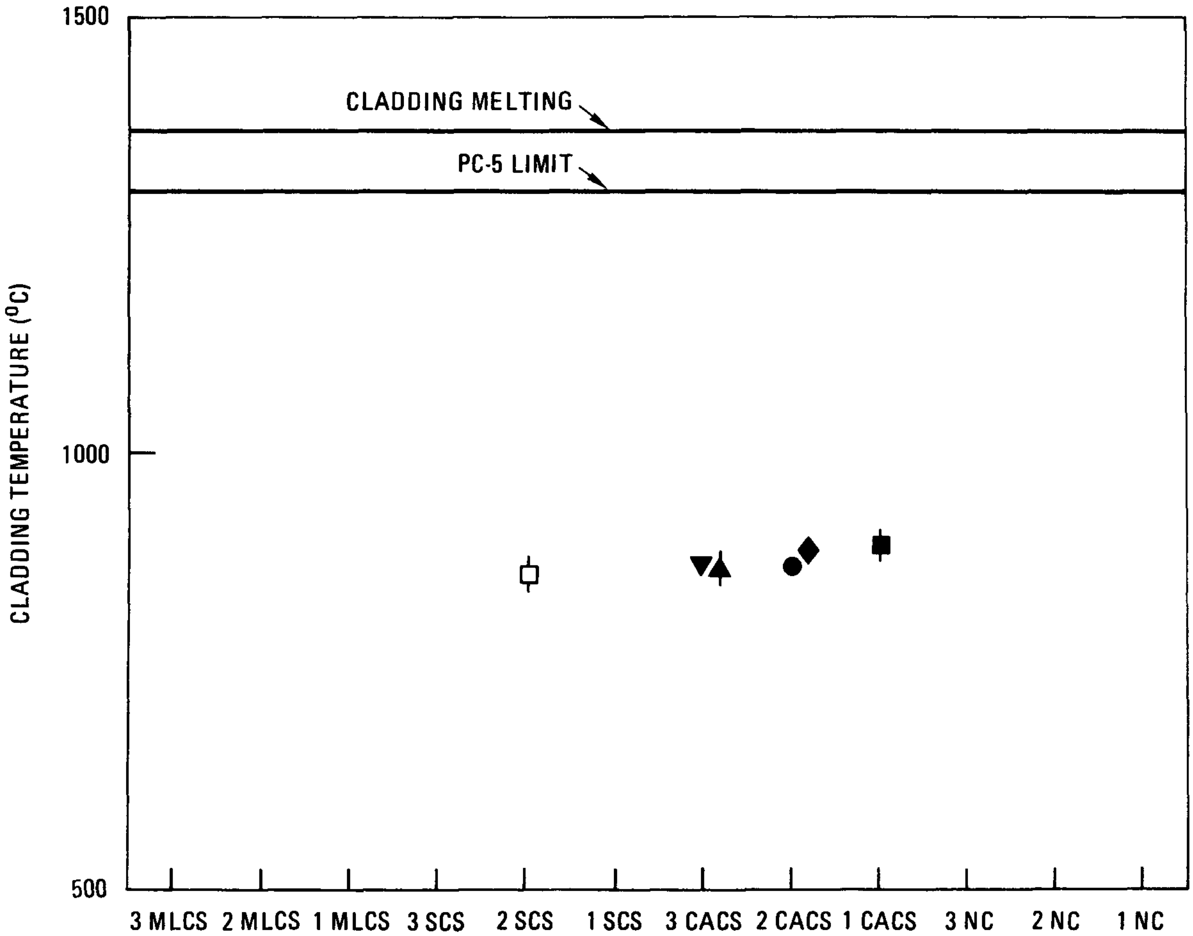

Fig. 5-20. Summary of core cooling performance in event category of decrease in reactor heat removal by secondary system; maximum blanket cladding temperature

5.4. DECREASE IN REACTOR COOLANT INVENTORY

Events which result in a decrease in reactor coolant inventory discussed in this section are the following:

1. Inadvertent opening of a PCRV pressure relief valve leading to a DBDA (Section 5.4.1).

2. Failure of small lines carrying primary coolant outside the PCRV (Section 5.4.2).

Case 1 is ANS PC-5. Case 2 is ANS PC-3.

To indicate the large margin in the GCFR core cooling capability, analyses were performed for cases using redundant RHR systems that are available, but not required, to meet the deterministic safety criteria discussed in Section 5.1.1. Section 5.4.4 summarizes these results.

5.4.1. Inadvertent Opening of a PCRV Pressure Relief Valve

5.4.1.1. Identification of Causes and Accident Description. An accidental depressurization of the reactor coolant system could result from an inadvertent opening of the PCRV pressure relief valve. The possible event sequences are as follows:

1. The relief valve lifts at a pressure less than its setpoint, the rupture disc bursts, and the relief valve reseats. This sequence of events is ANS PC-4. The consequence of this partial blowdown is minor and nonlimiting.

2. The relief valve lifts at a pressure less than its set pressure and fails to reseat, and the rupture disc does not burst. This sequence of events is ANS PC-5. Adequate operator alarm is provided, and design provision (e.g., a block valve) is available to

prevent an excessive release of primary coolant to the gas waste system. The consequence is nonlimiting.

3. The relief valve lifts at a pressure lower than its setpoint and fails to reseat or pressure relief pipe breaks, and the rupture disc bursts, leaving a conservative leak area up to 130-cm^2 (20-$in.^2$) open to the containment building. This accident is ANS PC-5.

Event sequence 3 above is the limiting case and is included in the category of DBDA, which is detailed below.

The DBDA is a conservative depressurization accident scenario in which the component design is based on achieving adequate core cooling. A gross failure of the PCRV or its closures is extremely improbable, since the PCRV is designed in accordance with American Society of Mechanical Engineers (ASME) Section III, Divisions 1 and 2. For this reason, failure of the LWR pressure vessels need not be postulated as a design basis event. However, in the GCFR, a leak area of up to 200 cm^2 (30 $in.^2$) is conservatively assumed.

The GCFR plant will be designed for this accident with the following conditions:

1. Containment pressure does not exceed design limit.

2. PCRV internals are capable of withstanding the pressure, flow, and temperature transients without failing in a way which would preclude adequate core cooling.

3. The back pressure in the PCRV (containment) is sufficient to ensure adequate cooling.

4. The transient pressure differentials will not cause levitation of the fuel elements or control rods.

5. The pressure and temperature transients do not cause a consequential moisture ingress event.

5.4.1.2. Sequence of Events and System Operation. A postulated break in the PCRV pressure boundary results in a rapidly decreasing reactor coolant pressure. The reactor is tripped by one of the PPS signals given in Tables 5-9 and 5-10. The following signals initiate the containment isolation actions:

1. Low PCRV pressure.
2. High containment atmosphere pressure.
3. High containment radioactivity.

For core cooling during a DBDA, the GCFR is equipped with two RHR systems, the MLCS, which includes some nonsafety components (see Section 4.5.1), and the CACS, which is an independent backup safety system (see Section 4.5.3). Two sequences of a DBDA event are reviewed: (1) where the MLCS is available and (2) where the MLCS is not available and the CACS is used for core cooling.

DBDA Sequence with MLCS. In conjunction with the reactor trip following a DBDA, the main circulator speed is reduced initially for shutdown operation. The limiting case of DBDA with MLCS is defined by assuming that one MLCS loop fails with its MLIV stuck open (see Appendix A). In this case, DBDA core cooling is performed with two MLCS loops and with significant core bypass flow through the failed loop.

The circulator has a speed control function which is inversely proportional to the coolant density. Because of this, the circulators of the two operating MLCS loops will automatically accelerate, trying to maintain constant mass flow rate for shutdown operation with decreasing coolant density until the speed levels out at the maximum design speed. The flow in the failed loop will coast down and reverse, bypassing the core through a stuck open MLIV. Essentially, the main circulators will be at full speed after complete blowdown (i.e., the primary coolant pressure equilibrates the containment pressure).

After the coolant blowdown, an air ingress passes into the primary coolant system through the leak area due to gas exchange between the containment and PCRV interior. The air ingress mechanisms are (1) thermally induced inhalation when the primary coolant contracts, (2) natural circulation between hot, light PCRV gas and cold, heavy containment gas, and (3) long-term molecular diffusion. The air ingress effect is accounted for in the DBDA transient analysis method. After substantial cooldown at low decay heat, the circulator speed will be run back manually for the long-term RHR.

At any time during MLCS RHR, if more than one MLCS loop fails due to an LOSP or other causes, the CACS will start up automatically. According to sensitivity studies, the later the transfer to CACS occurs, the less limiting the core cooling. It is due to a large MLCS RHR capability. Therefore, a failed MLCS due to LOSP at the initial stage of a DBDA is the most limiting case, as discussed below.

DBDA Sequence with CACS. In case of a DBDA and loss of the MLCS (e.g., LOSP), the main circulators will inertially coast down; during this period, the emergency diesel generators are started, and they will energize the safety RHR systems, SCS, or CACS with safety grade 1E power. A conservative diesel startup delay of 60 s is assumed. A DBDA is a design basis event for the CACS. When the main loop flow decays below the loop transfer threshold, the check valves will automatically switch in the auxiliary circulator flow. The main loop isolation valves (MLIVS) will close, and the auxiliary loop isolation valves (ALIVs) will open, permitting the core flow from auxiliary circulators. The CACS is a self-contained, independent system, which can provide continued RHR for an indefinite period.

5.4.1.3. Analysis of Effects and Consequences. The DBDA was analyzed for two aspects: (1) adequacy of core cooling and (2) integrity of the containment. Each aspect is described below.

The DBDA core cooling analysis was performed using the system dynamics program FASTRAN (Section 5.1.6.1). The design basis leak area of 200 cm^2 (30 $in.^2$) is assumed conservatively to occur at the reactor inlet plenum. The leak flow is choked during most of the depressurization transient. When it is unchoked at the end of the blowdown, a full contraction loss and a full expansion loss are assumed upstream and downstream of the leak area, whereas other frictional losses in the leak flow passage are neglected. Also, the conservative depressurization model is assumed by allowing all the system parameter uncertainty margins, as shown in Table 5-8.

The plant control system is assumed to be in the automatic mode.

DBDA Core Cooling with MLCS. If the MLCS is available, the main circulator speed, initially reduced, will automatically increase to maintain the same core mass flow rate with decreaseing coolant density until it levels out at the maximum design speed [see Fig. 5-21(e)]. Figure 5-21(a) illustrates the nuclear power, the coolant pressure, and the core flow transients following depressurization. The coolant flow rate decreases initially as the circulator speed is reduced, and the core flow transient in Fig. 5-21(a) shows rapid decrease after reactor trip and another sharp decrease at laminar core transition. The flow transient also indicates a gradual increase in mass flow rate due to air ingress following a complete blow-down. Figure 5-21(c) shows the transient response of the hot spot cladding temperatures of the maximum powered fuel and blanket rods. Both rods reach maximum temperatures that are lower than the respective design temperature limits for PC-5 (see Section 5.1.2).

DBDA Core Cooling with CACS. With assumed LOSP, the MLCS is disabled, the main circulators coast down, and the emergency diesel generators are started to provide 1E power for the CACS. The RHR initiation system (Section 5.1.5.2) selects the SCS, and the pony motors are powered to prevent the main circulator speed from coasting down excessively and to maintain a prescribed value for the SCS operation. As pressure decays, the pony motor accelerates, automatically trying to provide the same mass flow rate until the maximum pony motor speed is reached.

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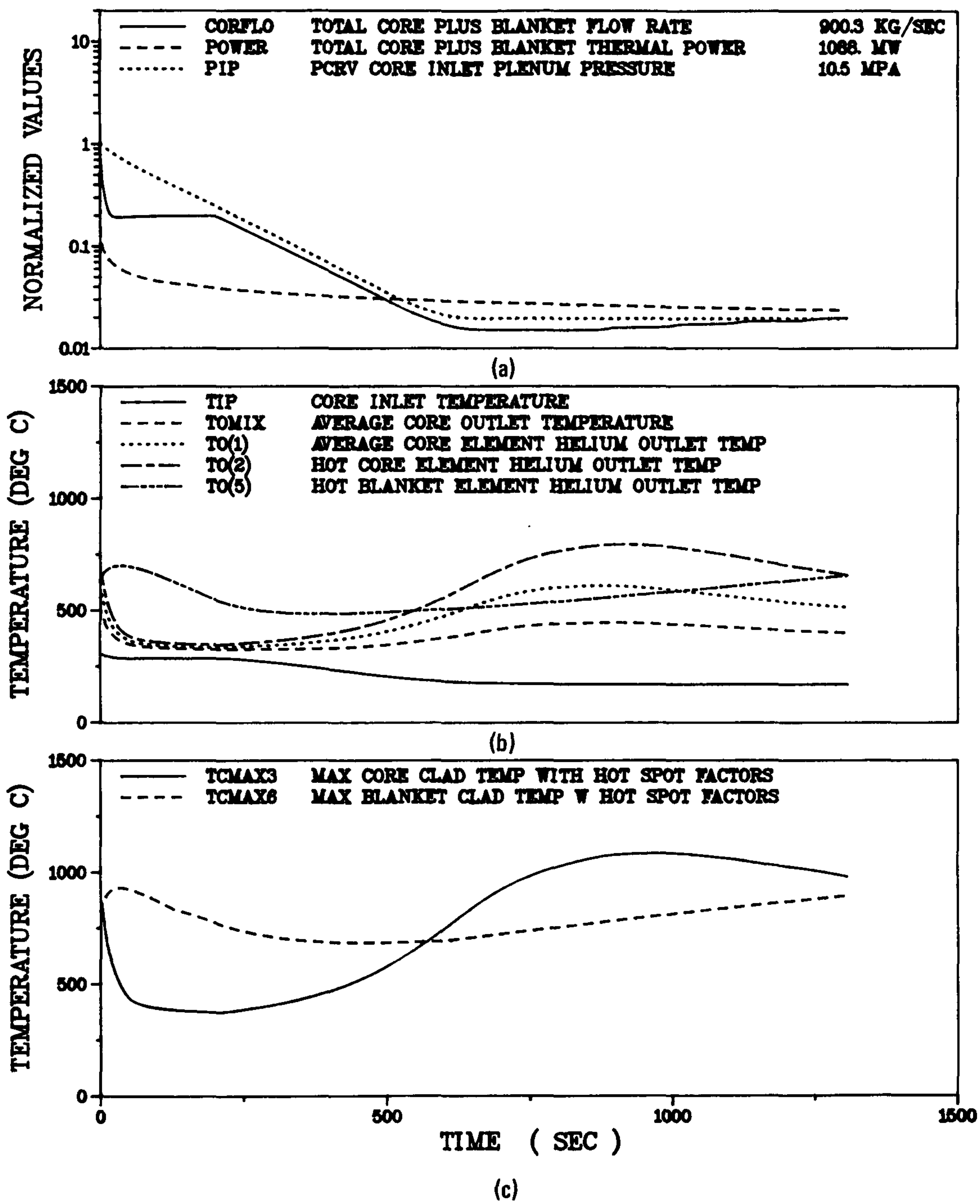

Fig. 5-21. DBDA using two MLCS and with MLIV stuck open: (a) core power, flow, and pressure, (b) inlet and outlet gas temperatures, (c) maximum cladding temperatures (sheet 1 of 2)

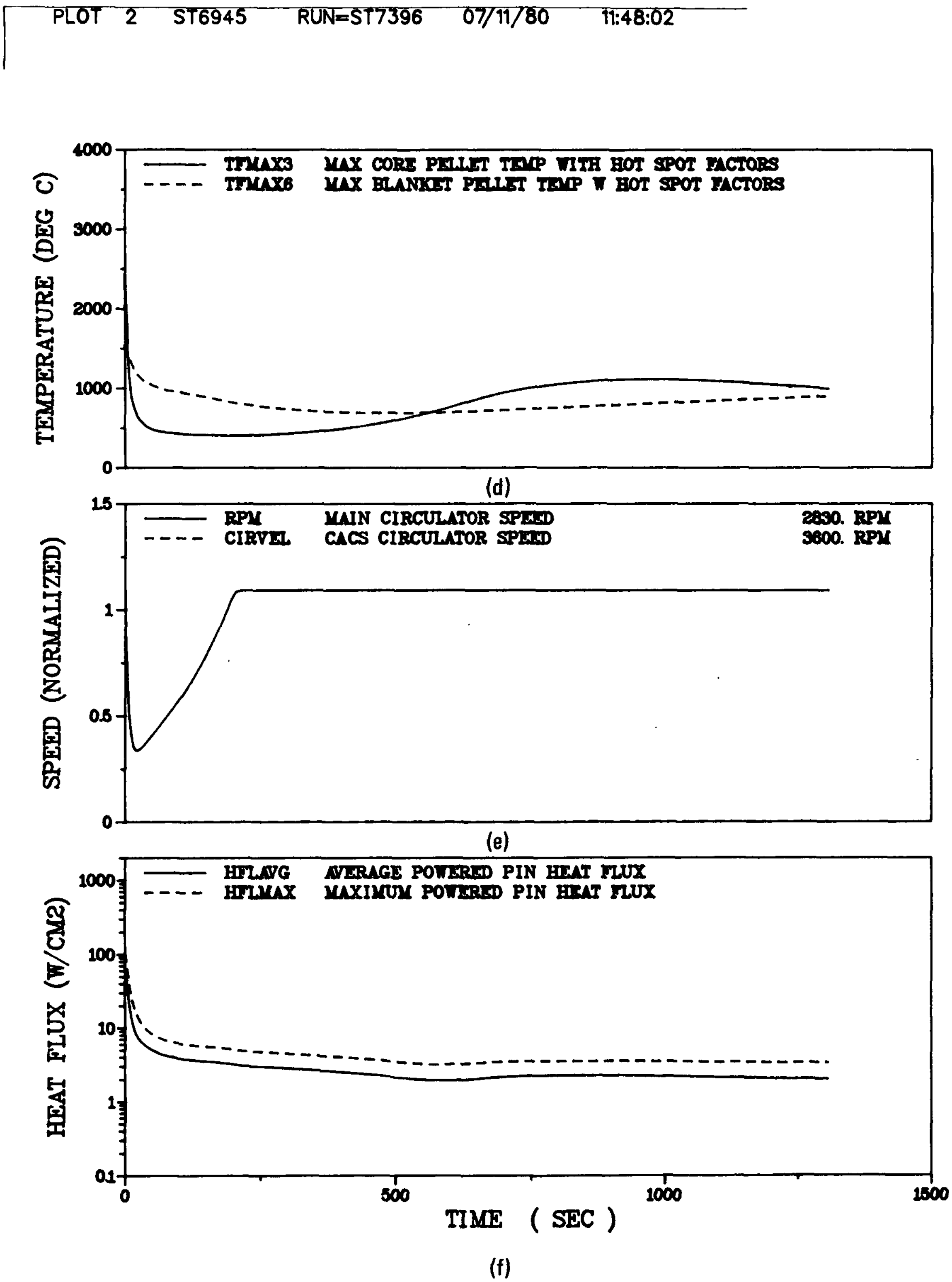

Fig. 5-21. DBDA using two MLCS and with MLIV stuck open: (d) maximum fuel temperature, (e) circulator speeds, (f) cladding heat flux (sheet 2 of 2)

When the primary coolant pressure decreases below 2.07 MPa (300 psia), the RHR initiation system (Section 4.5.3.3) recognizes depressurization with insufficient MLCS flow and initiates the CACS. The circulator pressure head forces the auxiliary loop isolation valve (ALIV) open and the MLIV closed. When the CACS is operationsl, the SCS is shut down.

The single failure criterion may be applied, assuming failure of one CACS loop or failure of one MLIV to close after transfer to the CACS mode. The former case is more limiting for the core cooling and is analyzed herein.

Figure 5-22 shows the system response to the DBDA. Figure 5-22(a) illustrates the nuclear power, coolant pressure, and core flow transients following the depressurization. Nuclear power is maintained at the initial value until reactor trip occurs on high power-to-flow ratio at ~2.2 s.

The core flow decreases rapidly after reactor trip and the main circulator trip. The primary coolant pressure reaches 2.07 MPa (300 psia) at 333 s, when the auxiliary circulators are turned on. The auxiliary loop flow replaces the main loop core flow at 344 s. Figure 5-22(e) shows the main circulator (RPM) and the auxiliary circulator rotative speed (CIRVEL) variation during this transient. Figure 5-22(e) shows that the main circulators coast down until the auxiliary circulators accelerate and replace the coolant circulating function. As the coolant pressure decays further, the auxiliary circulator speed is increased until the blowdown is complete.

Figure 5-22(f) shows the transient core heat fluxes for the hot spot (HFLMAX) and the average (HFLAVG) fuel cladding surfaces. Figure 5-22(c) shows the transient response of the hot spot cladding temperatures of the maximum-powered fuel (TCMAX3) and blanket (TCMAX6) rods. Both rods reach maximum tmeperatures that are lower than their respective design temperature limits for the PC-5 (see Section 5.1.2). Figure 5-22(d) shows the maximum fuel temperature transient. Figure 5-22(b) shows the primary coolant inlet (TIP), reactor mixed outlet (TOMIX), fuel high-power channel outlet [TO(2)],

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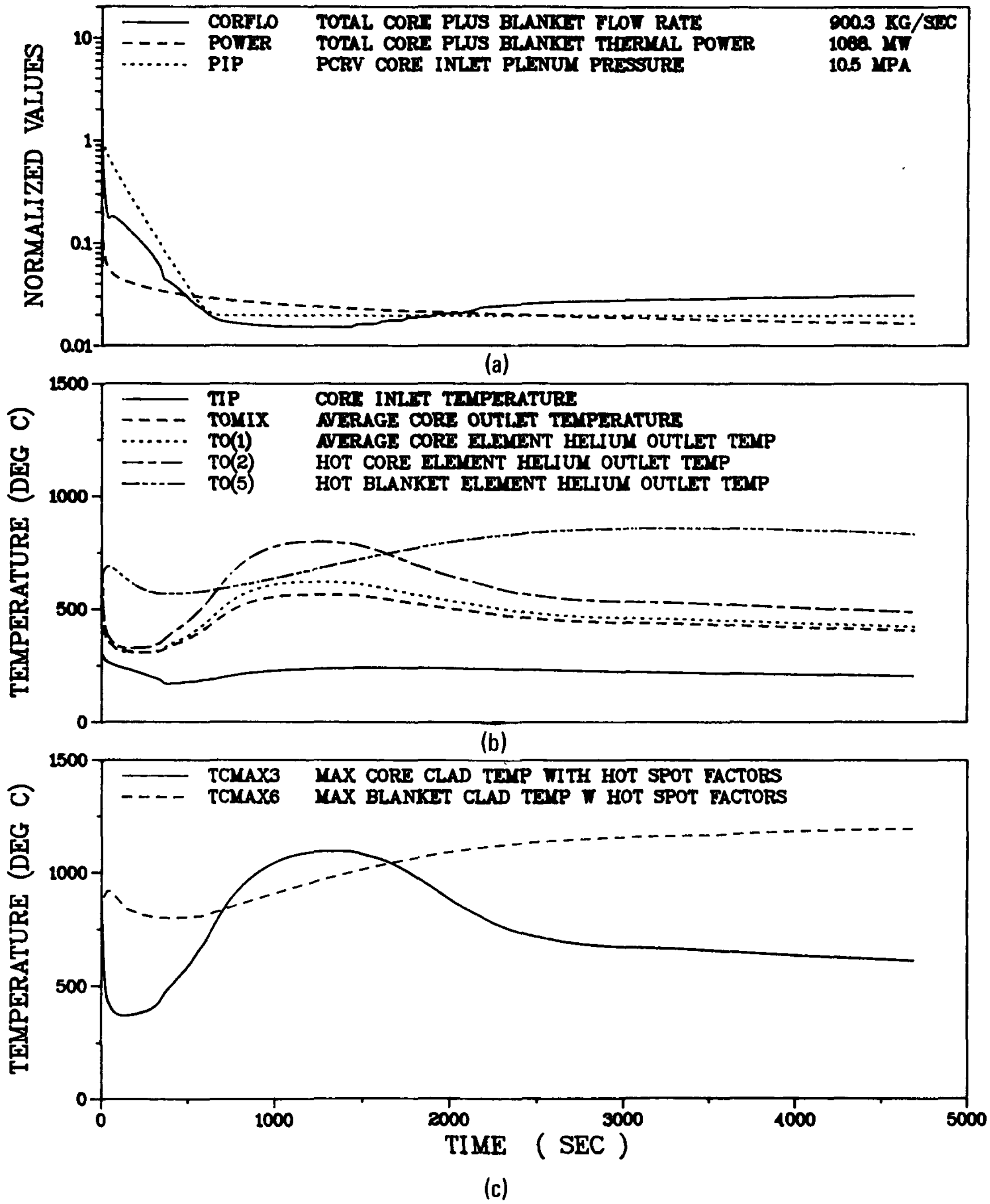
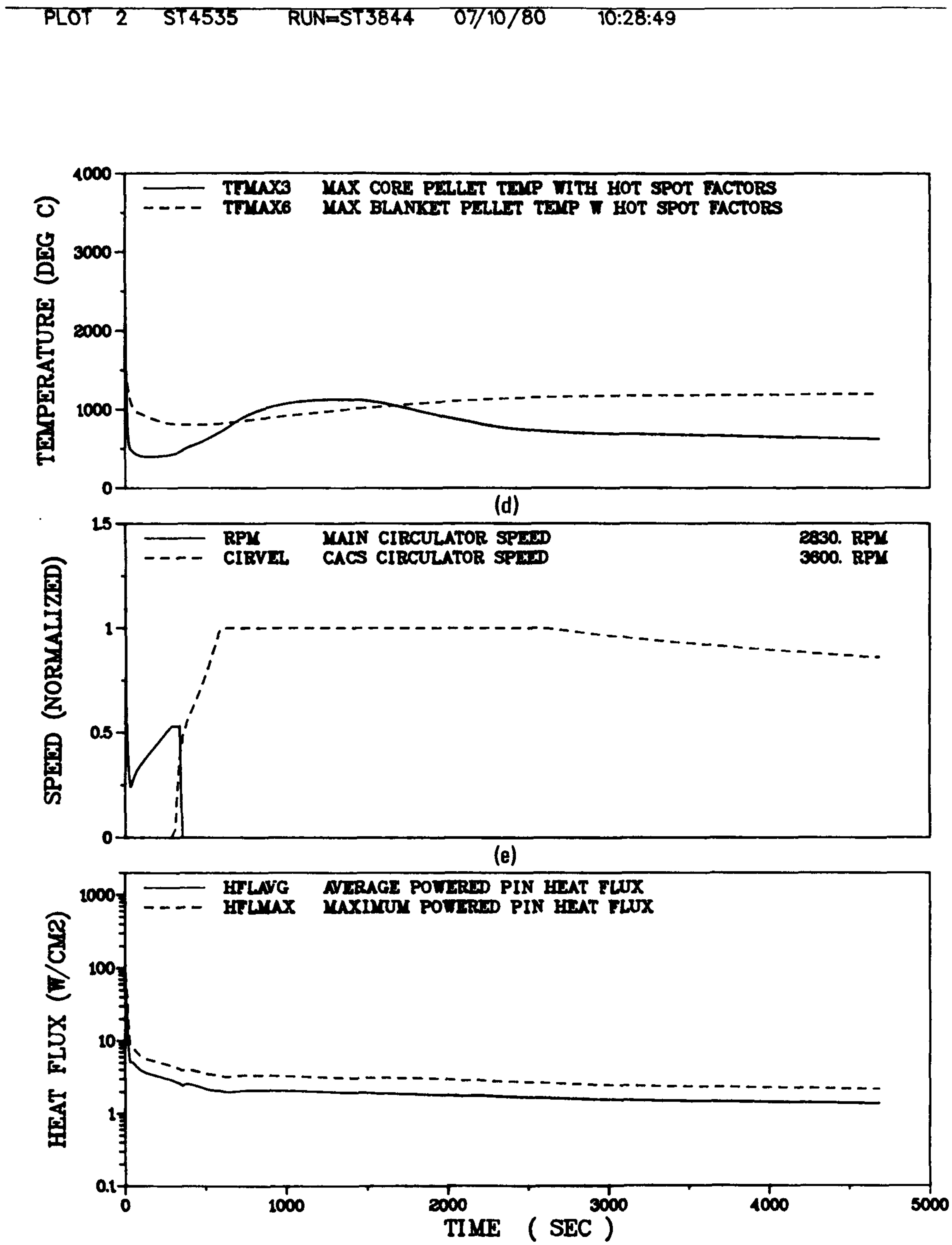


Fig. 5-22. DBDA using two CACS loops: (a) core power, flow, and pressure, (b) inlet and outlet temperatures, (c) maximum cladding temperatures (sheet 1 of 5)



(f)

Fig. 5-22. DBDA using two CACS loops: (d) maximum fuel temperature, (e) circulator speeds, (f) cladding heat flux (sheet 2 of 5)

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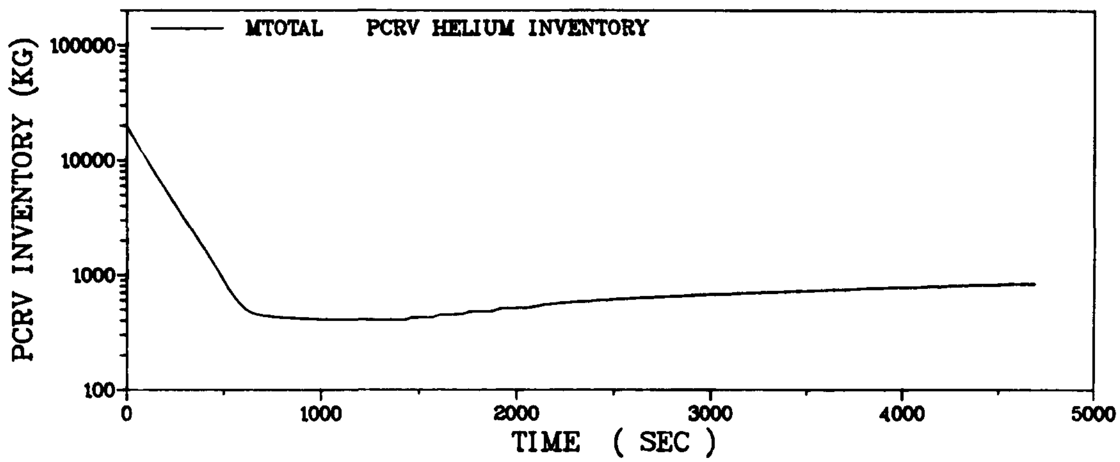

(g)

Fig. 5-22. DBDA using two CACS loops: (g) primary coolant inventory (sheet 3 of 5)

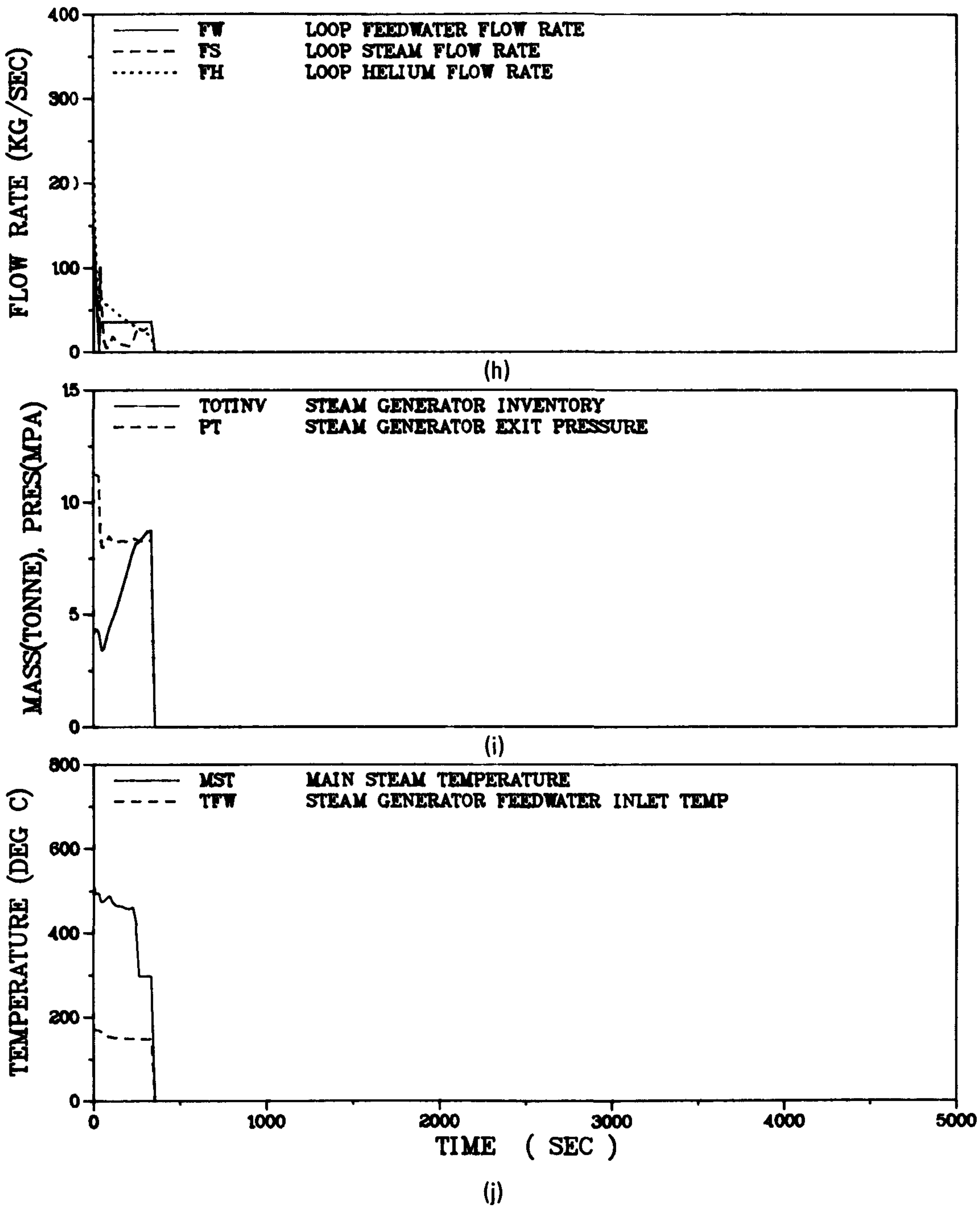

Fig. 5-22. DBDA using two CACS loops: (h) steam generator flow rates, (i) steam generator inventory and pressure, (j) steam generator temperatures (sheet 4 of 5)

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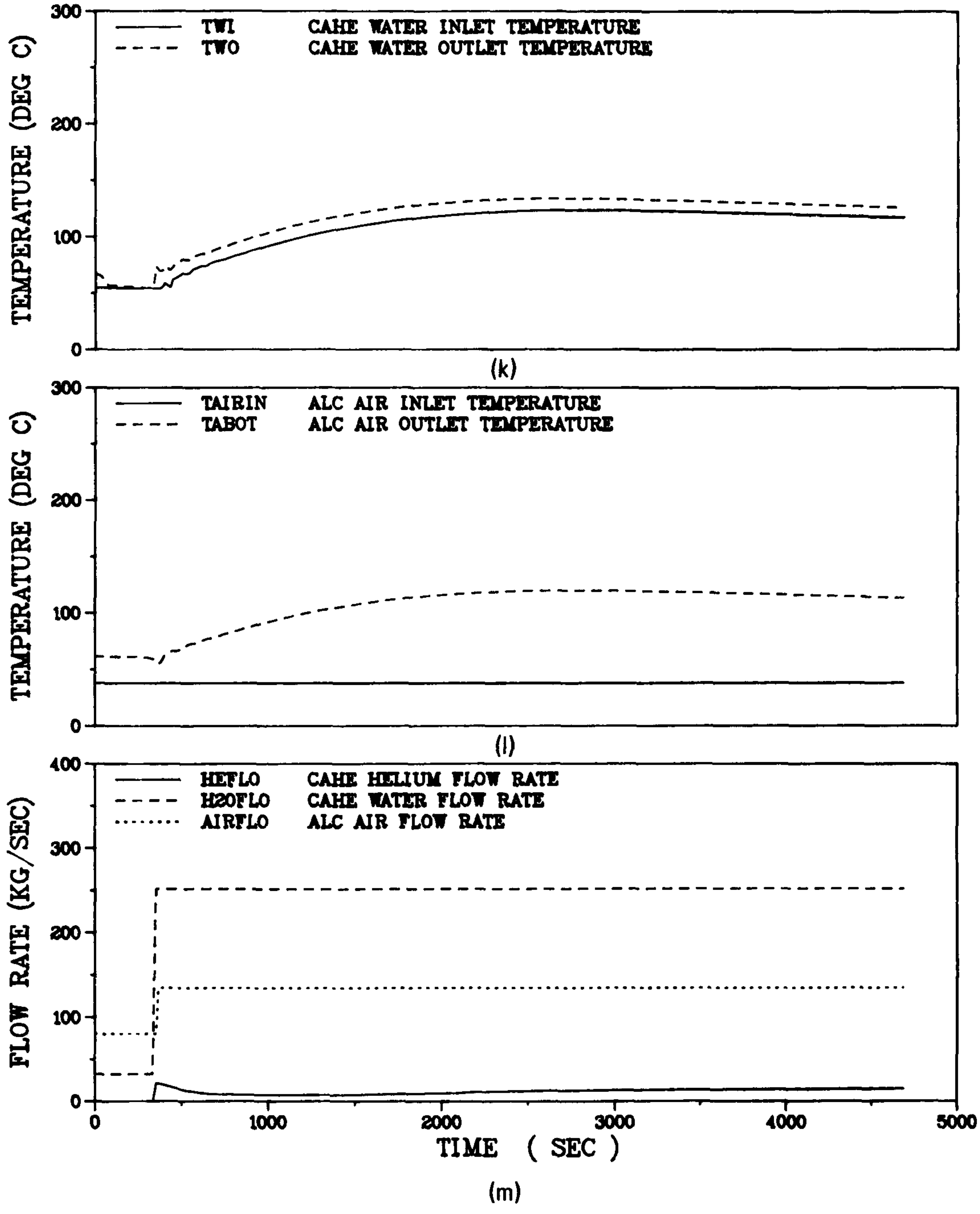

Fig. 5-22. DBDA using two CACS loops: (k) CAHE fluid temperatures, (l) ALC fluid temperatures, (m) CAHE and ALC flow rates (sheet 5 of 5)

and blanket high-power channel outlet [TO(5)]. The PC-5 component design limits, particularly the circulator, steam generator, and thermal barrier (Section 5.1.2), are adequately satisfied by the hot (TOMIX) and cold (TIP) leg maximum gas temperatures.

Figure 5-22(g) shows the coolant inventory change. The blowdown is completed at about 11 min. Figure 5-22(h,i,j) shows the secondary coolant conditions in the steam generators prior to transfer to the CACS.

Figure 5-22(m) summarizes the flow transients for the CACWS, namely the water flow rate (H2OFLO), primary coolant flow rate, and air flow rate in the cooling tower. Figure 5-22(k) shows the water temperatures for the inlet (TWI) and the outlet (TWO) of the CAHE. The CAHE water outlet temperature increases rapidly after the loop transfer to CACS. The water inlet temperature responds with a time lag, indicating the CACWS circuit transit time. The maximum water temperature indicates an adequate margin [160°C (290°F)] to the boiling point of 9 MPa (1300 psia) water. Figure 5-22(l) shows the air inlet (TAIRIN) and outlet (TABOT) temperatures in the air cooling tower.

Containment Response to DBDA. The containment response to a DBDA was analyzed using computer program CNTB (see Section 5.1.6.3). Two types of containment analyses were performed. The first analysis used a set of assumptions that minimize the containment pressure (i.e., the back pressure for the reactor coolant) to conservatively determine core cooling adequacy. The second analysis used an alternative set of assumptions which maximize the containment pressure to conservatively ascertain containment structural integrity. Table 5-13 shows these assumptions for the two types of analysis.

TABLE 5-13
ASSUMPTIONS USED FOR CONTAINMENT RESPONSE ANALYSES

	Analysis with Minimized Back Pressure	Analysis with Maximized Back Pressure
Location of leak	Core inlet plenum	Core outlet plenum
Containment Gas Mixing	Perfect mixing	No gas mixing

FASTRAN input and output for transient leak flow rates use the CNTB output for the containment temperature and pressure transients. CNTB input uses PCRV gas temperatures for FASTRAN output. The coupled results were obtained by a few iterations between the CNTB and FASTRAN calculations.

Curve A in Figs. 5-23 and 5-24 shows the pressure and temperature responses, respectively, of the containment atmosphere following a DBDA in which a set of assumptions was made to minimize the containment pressure.

Curve B in Fig. 5-23 and curves B1 and B2 in Fig. 5-24 show the pressure and temperature responses, respectively, which are calculated using a set of assumptions that maximizes the containment pressure. The peak pressure of 0.33 MPa (47 psia) obtained here indicates significant margins to the tentative design values of 0.41 MPa (60 psia) for the containment building. In Fig. 5-23, the unmixed containment gas temperatures are shown by curves B1 and B2 for helium and air, respectively. Although the postulated helium pocket temperature peak is high, the containment wall temperature peak is only 85°C (185°F), compared with the tentative structural design temperature of 155°C (311°F). The containment structural integrity is concluded to be assured during the postulated DBDA.

Figure 5-23 shows that, even in the case of the minimized back pressure, the transient values are significantly higher than the equilibrium back pressure, which is 0.25 MPa (36 psia). This high transient back pressure is a safety margin which has not been utilized, since all the depressurization core cooling analyses in this report (except for the coupled FASTRAN analysis in this section) are based on the constant equilibrium back pressure with a conservative factor. A significant core cooling advantage would be expected if the transient back pressure is applied.

5.4.1.4. Radiological Consequences. The radiological consequences of this event are negligible, since no fuel damage is predicted and correct containment isolation is assumed. The containment cleanup system removes any significant radioactivity in the released gas.

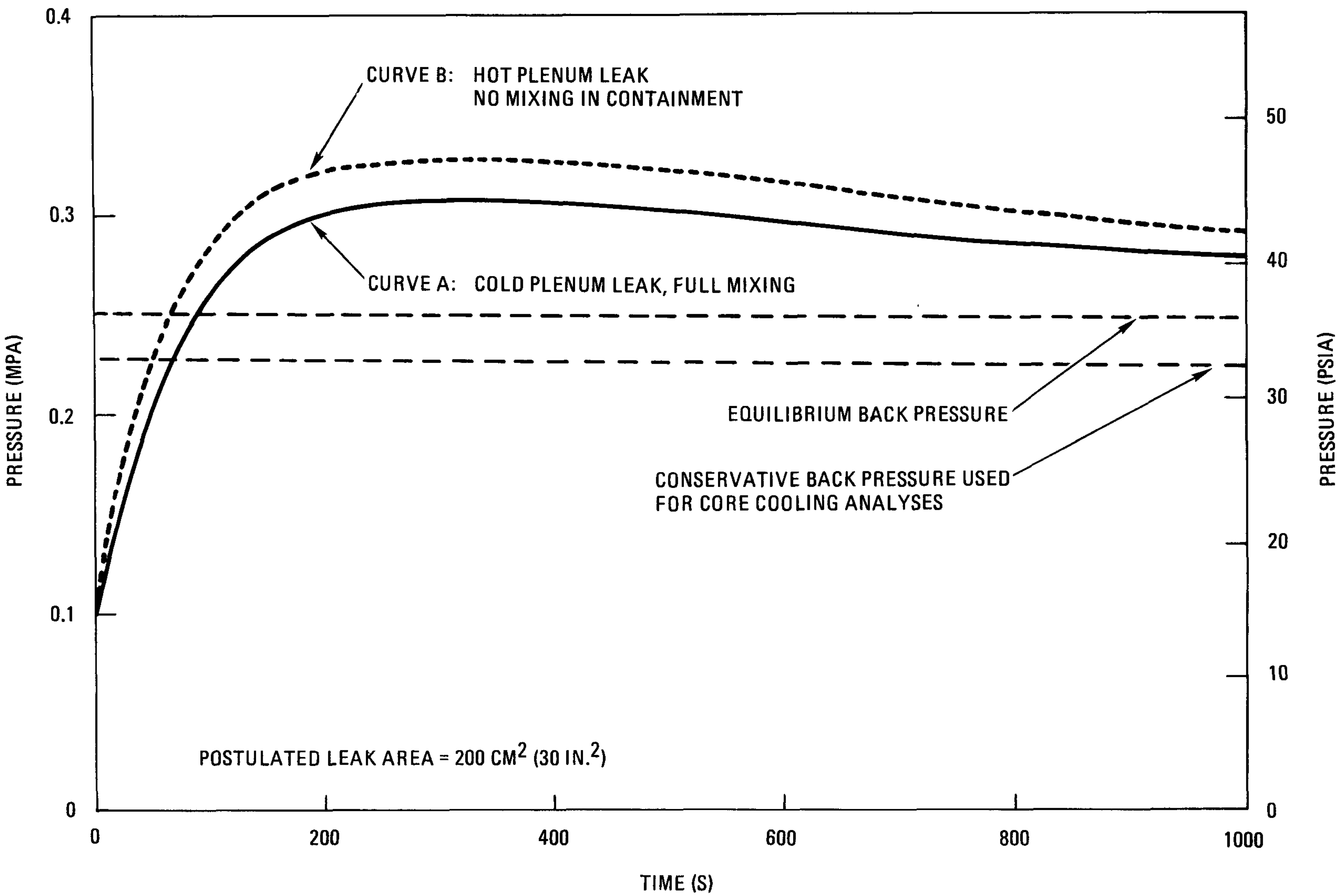

Fig. 5-23. Containment building transient following a DBDA, time versus pressure

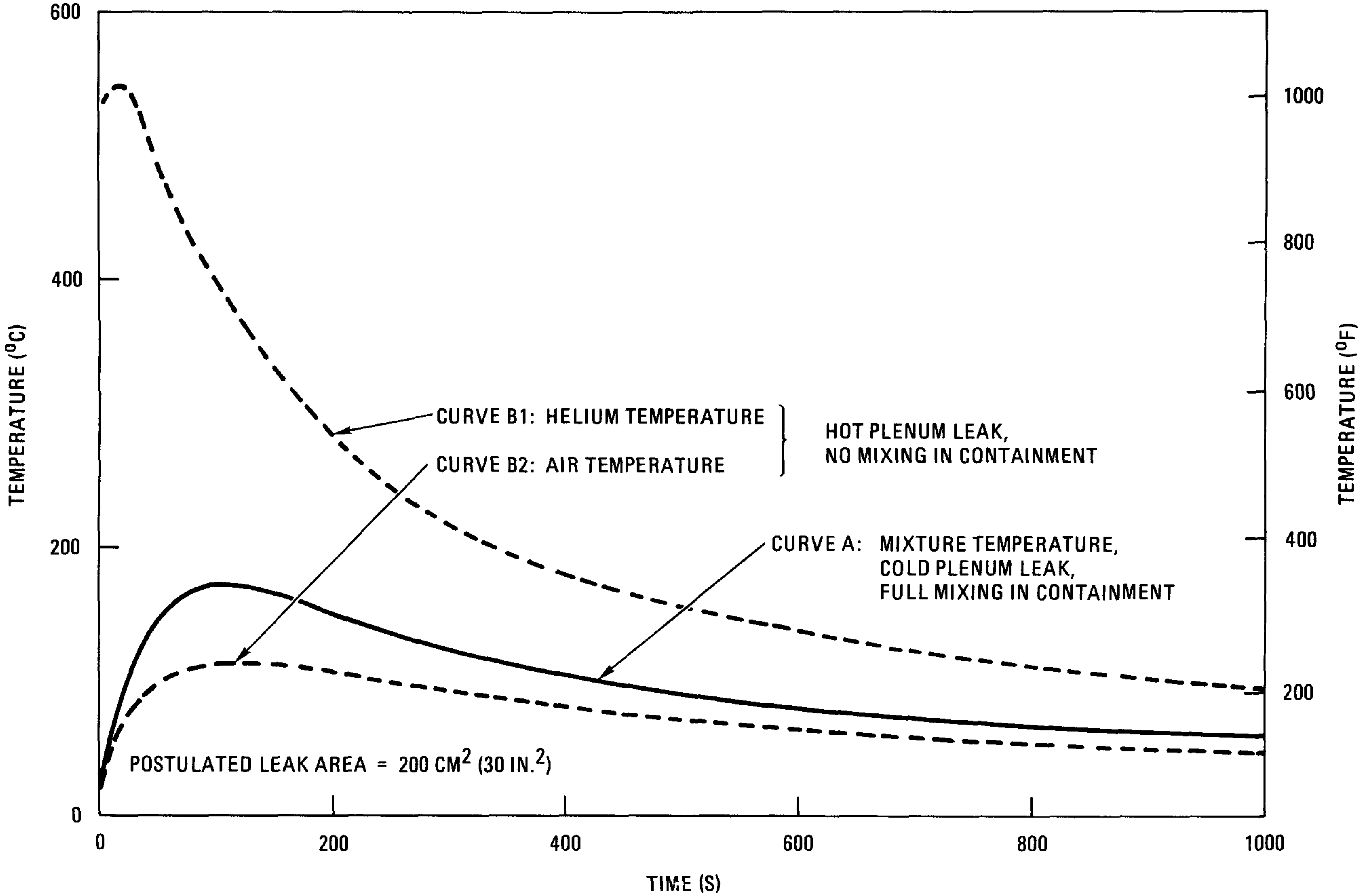

Fig. 5-24. Containment building transient following a DBDA, time versus temperature

5.4.1.5. Conclusions. The reactor system analysis shows that the PPS signals and the CACS core cooling system adequately protect against the DBDA. The maximum fuel and blanket rod cladding, and the primary coolant temperatures are maintained below those of the PC-5 design limit in Table 5-4. The containment response analysis indicates that the transient peak pressure and temperatures during the DBDA are significant. The containment building will be designed adequately, so that the structural integrity of the containment is maintained during the DBDA.

5.4.2. Failure of Small Lines Carrying Primary Coolant Outside PCRV

5.4.2.1. Identification of Causes and Accident Description. The accident results from a break in small lines [6.5 cm^2 (1 $in.^2$)], such as an instrument line connected to the primary coolant system. This accident results in a slow depressurization. This case is ANS PC-3. Upon leak detection prior to a PPS trip, the operator takes appropriate action and terminates the leak or shuts down the plant. The shutdown core cooling can be performed by the MLCS, SCS, or CACS.

5.4.2.2. Analysis of Effects and Consequences. Since the operator detects the leak and takes appropriate actions to terminate the accident, no significant consequences to the reactor or its essential auxiliary systems result.

In case of no operator action, the reactor will be tripped at ~500 s on low coolant pressure PPS. At some point in the blowdown period, containment isolation will occur due to a signal of either low primary coolant pressure or high pressure or radioactivity in the containment. In the safety-related event sequence, LOSP and loss of one cooling loop as single failure are assumed. Consequently, two SCS loops are assumed to perform the core cooling. The main circulator speed is sharply reduced after the reactor trip. The SCS pony motor prevents the main circulator from coasting down below the set speed (30%) and accelerates automatically as coolant pressure and density decay until the maximum pony motor speed (50%) is reached. When the system pressure reaches 2.07 MPa (300 psia) in ~2 h, the RHR initiation system recognizes depressurization and switches on the CACS, as described in the previous section. The PCRV takes ~4 h to fully depressurize with a leak area of 6.5 cm^2 (1 $in.^2$). The core decay heat rate is ~1% at this time. The core fuel and the blanket cladding and the primary coolant temperatures are maintained well below the respective limits for the PC-3 (Table 5-4) during the SCS or the CACS operation.

5.4.2.3. Radiological Consequences. The plant design has no unusual features which would prevent limiting the radiological consequences to an acceptable level if a small line carrying primary coolant outside the PCRV fails. Reactor coolant activity concentrations, isolation valve closing time, and leak rates would be appropriately limited.

5.4.3. Margin Cases Beyond Deterministic Safety Criteria

To demonstrate a large margin in the GCFR RHR capability, the following sections summarize cases using redundant RHR systems that are available, but not required, for safety-related actions based on the deterministic rules.

5.4.3.1. DBDA Core Cooling with One MLCS Loop. Due to large MLCS capability, adequate core cooling can be provided with only one MLCS loop under the DBDA conditions. This is a case of multiple failures beyond the deterministic safety evaluation requirement (see Fig. 5-25). Figure 5-25(a) shows the transient responses of the core flow, power, and inlet pressure. Figure 5-25(b) shows the coolant temperature responses at various locations. Figure 5-25(c) shows the maximum fuel and blanket cladding temperatures. The hot spot cladding temperature of a typical fuel rod in the maximum-powered assembly is shown to be 867°C (1593°F), which is well below the PC-5 limit, indicating adequate margin for RHR performance.

5.4.3.2. Depressurization with a Large Leak Area. The PCRV cavity closures employ flow restrictors which will limit the leak rate with a flow area less than design basis value, 200 cm^2 (30 $in.^2$) if the closure seal fails. To demonstrate a large margin of the core cooling capability, a rapid depressurization is assumed for a 650-cm^2 (100-$in.^2$) leak area.

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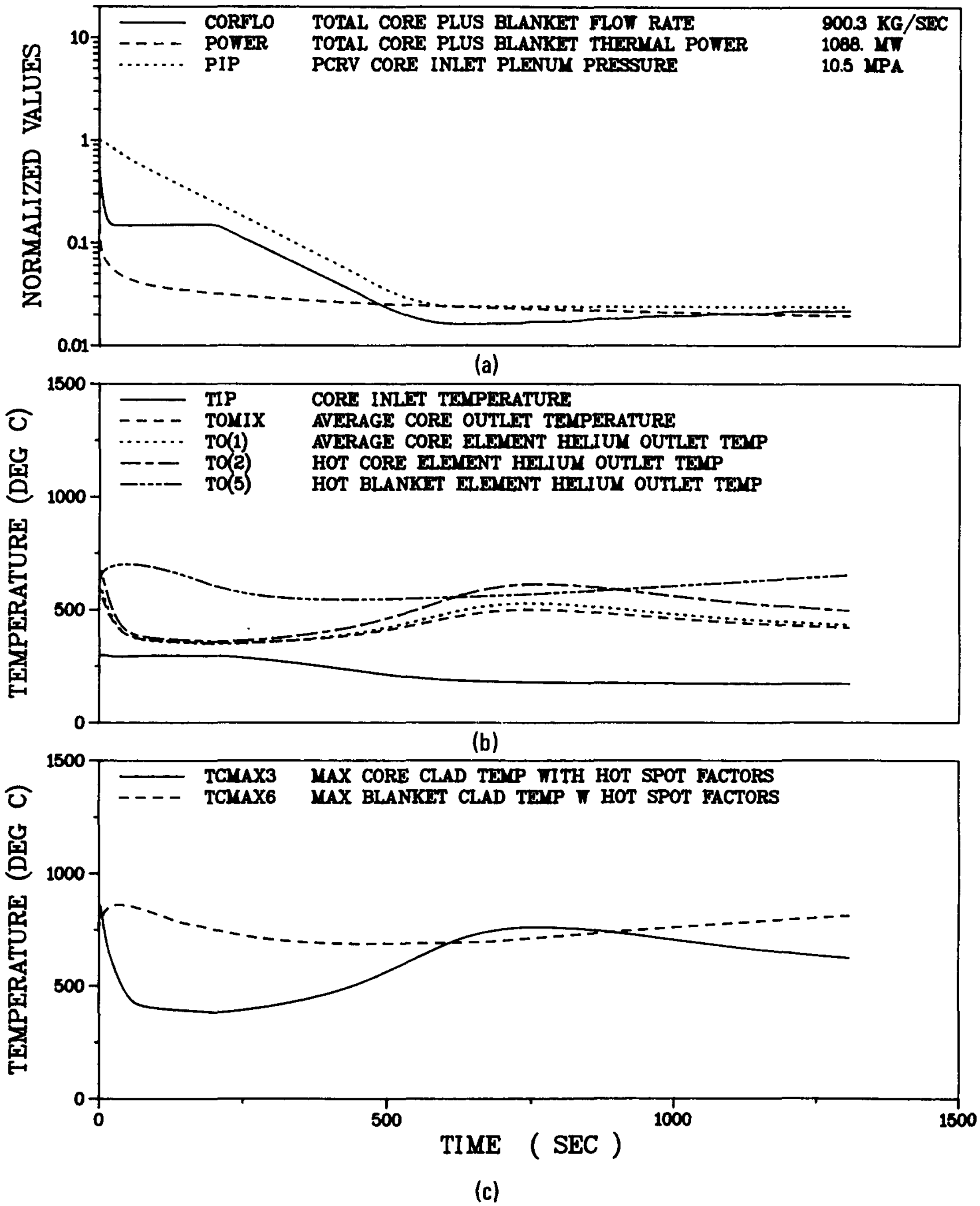

Fig. 5-25. DBDA using one MLCS loop: (a) core power, flow, and pressure, (b) inlet and outlet gas temperatures, (c) maximum cladding temperatures

Three MLCS loops are used for this case. Figure 5-26(a,b,c) shows results of the system parameter response. The maximum fuel and blanket cladding temperatures are 917° and 1047°C (1683° and 1917°F), respectively, which are significantly lower than the PC-5 limit [1300°C (2372°F)], indicating adequate RHR in this margin case.

5.4.3.3. DBDA with Natural Circulation Heat Sink. Natural circulation in primary coolant loops under depressurized condition is not effective for adequate RHR unless repressurization is available (see Section 5.6.3). However, natural circulation backup in the secondary water loops and the tertiary air cooling tower is always available and adequate for RHR in case of forced circulation by the pumps and the fans failing following a DBDA. Figure 5-27(a through f) shows system parameter response during the DBDA core cooling with two CACS loops having forced circulation in helium and natural circulation in the water and the air flows. The maximum fuel and blanket cladding temperatures are 1127° and 1260°C (2061° and 2300°F), respectively, which meet the PC-5 limit of 1300°C (2372°F). The water temperature response in Fig. 5-27(d) indicates adequate margin to 300°C (577°F) boiling temperature for 9 MPa (1300 psia) water.

5.4.4. Summary and Conclusion for Decrease in Reactor Coolant Inventory

To provide a perspective for depth of protection provided by the RHR systems, Figs. 5-28 and 5-29 summarize the results of all cases of this event category, including the margin cases. Figures 5-28 and 5-29 show maximum fuel and blanket cladding temperatures, respectively, with an abscissa indicating number and type of the RHR system loops used. Dark symbols signify the margin cases which assume multiple failures beyond the deterministic safety requirements.

These figures indicate adequate core cooling performance and significant depth of redundancy in available backup RHR systems.

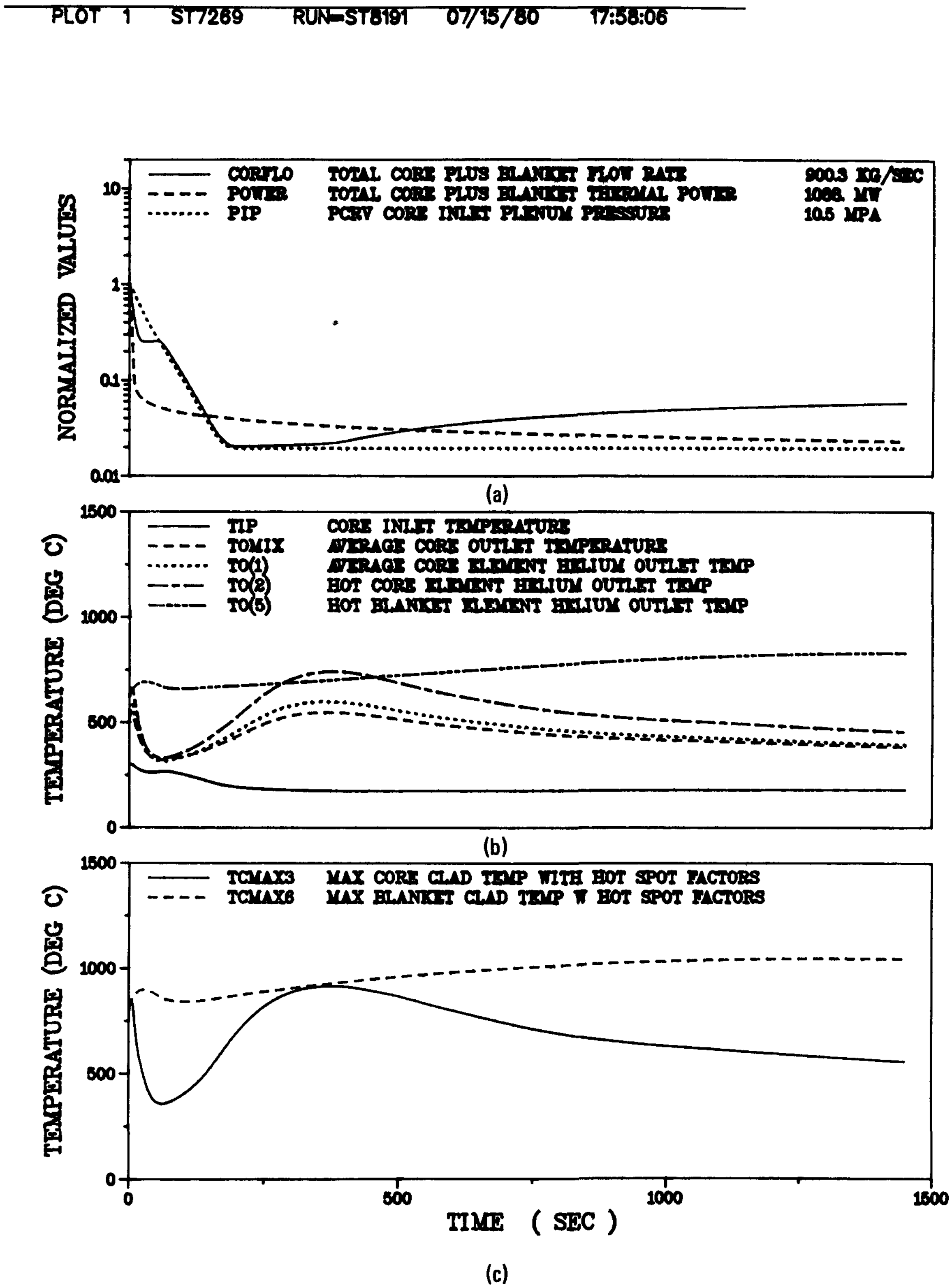

Fig. 5-26. DBDA without leak flow restrictor (650-cm^2 leak area): (a) core power, flow, and pressure, (b) inlet and outlet gas temperatures, (c) maximum cladding temperatures

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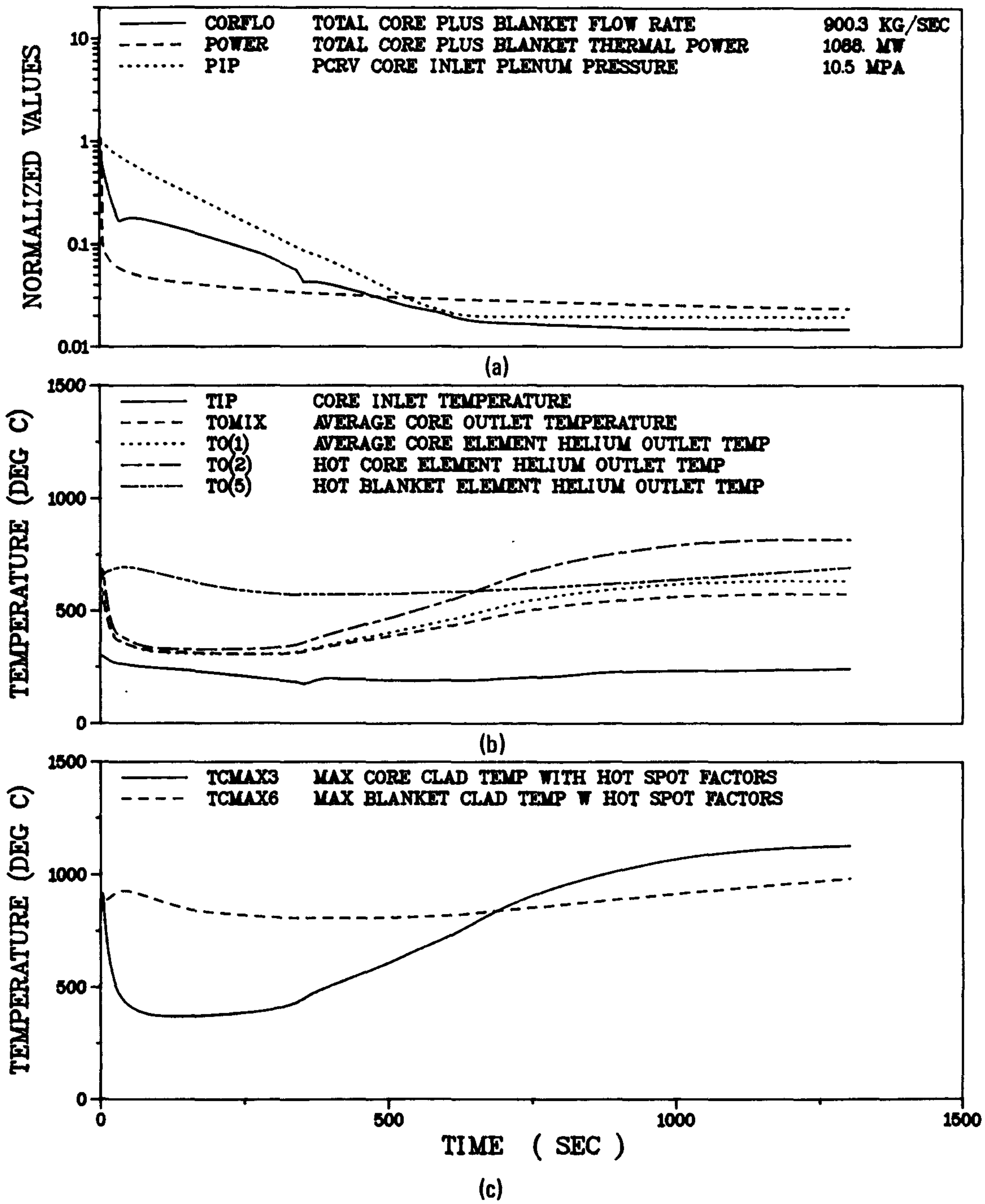

Fig. 5-27. DBDA with natural circulation heat sink in CACWS: (a) core power, flow, and pressure, (b) inlet and outlet gas temperatures, (c) maximum cladding temperatures (sheet 1 of 2)

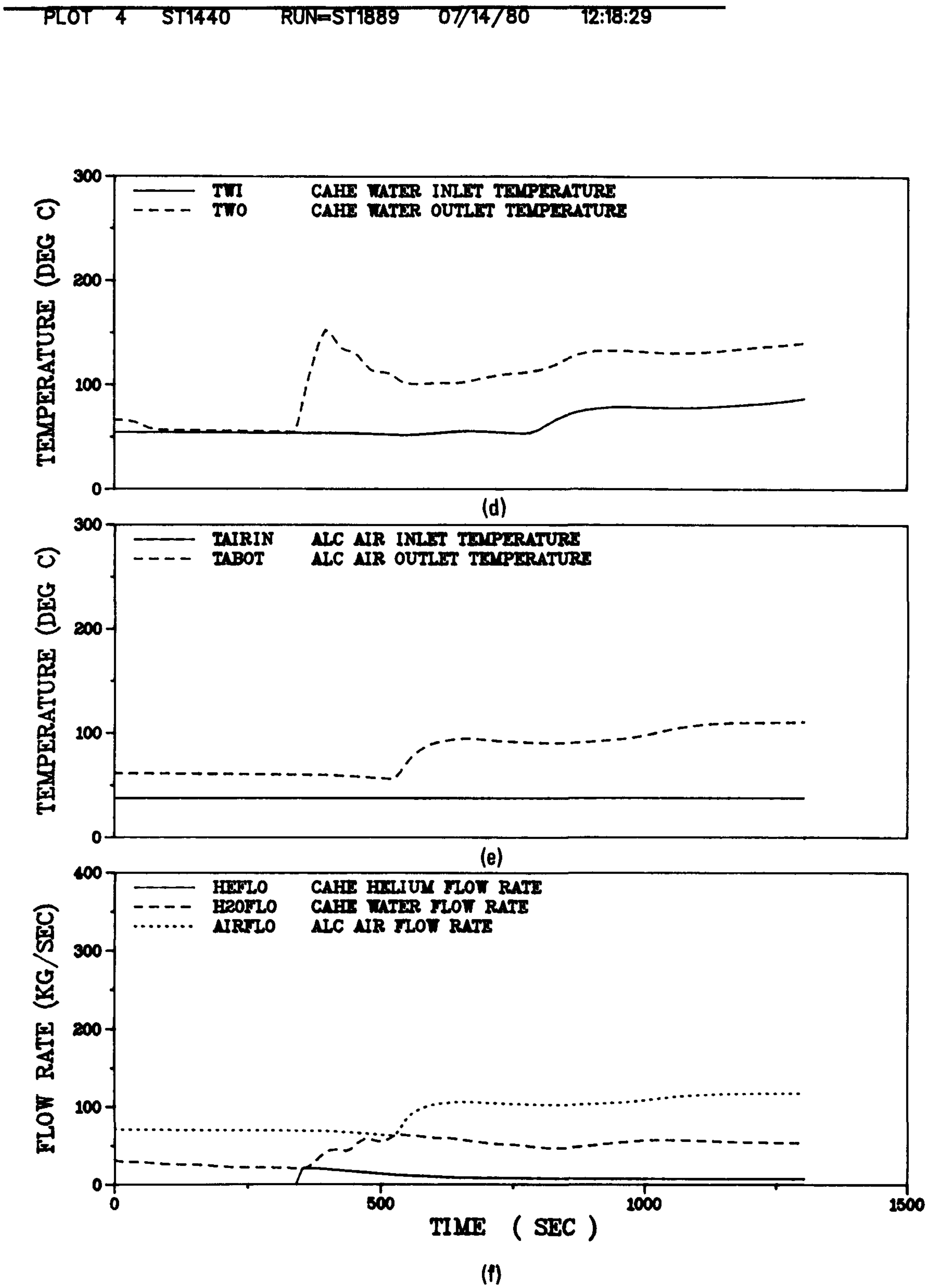

Fig. 5-27. DBDA with natural circulation heat sink in CACWS: (d) CAHE fluid temperature, (e) ALC fluid temperature, (f) CAHE and ALC flow rates (sheet 2 of 2)

SYMBOL	INITIATING EVENT	ADDITIONAL FAILURES	MAX CLADDING TEMPERATURE (^oC)	LIMIT TEMP (^oC)	PLANT CONDITION
⌓	DBDA	0	729	1300	PC-5
D	DBDA	1	898	1300	PC-5
ᑫ	DBDA	2	1086	1300	PC-5
◓	DBDA	2	867	1300	BEYOND PC-5
◗	DBA WITH A LARGE LEAK AREA	1	917	1300	BEYOND PC-5
◡	DBDA WITH LOSP	4	1097	1300	PC-5
⏀	DBDA WITH LOSP	4	826	1300	PC-5
◖	DBDA WITH LOSP AND NC IN HEAT SINK	6	1127	1300	BEYOND PC-5

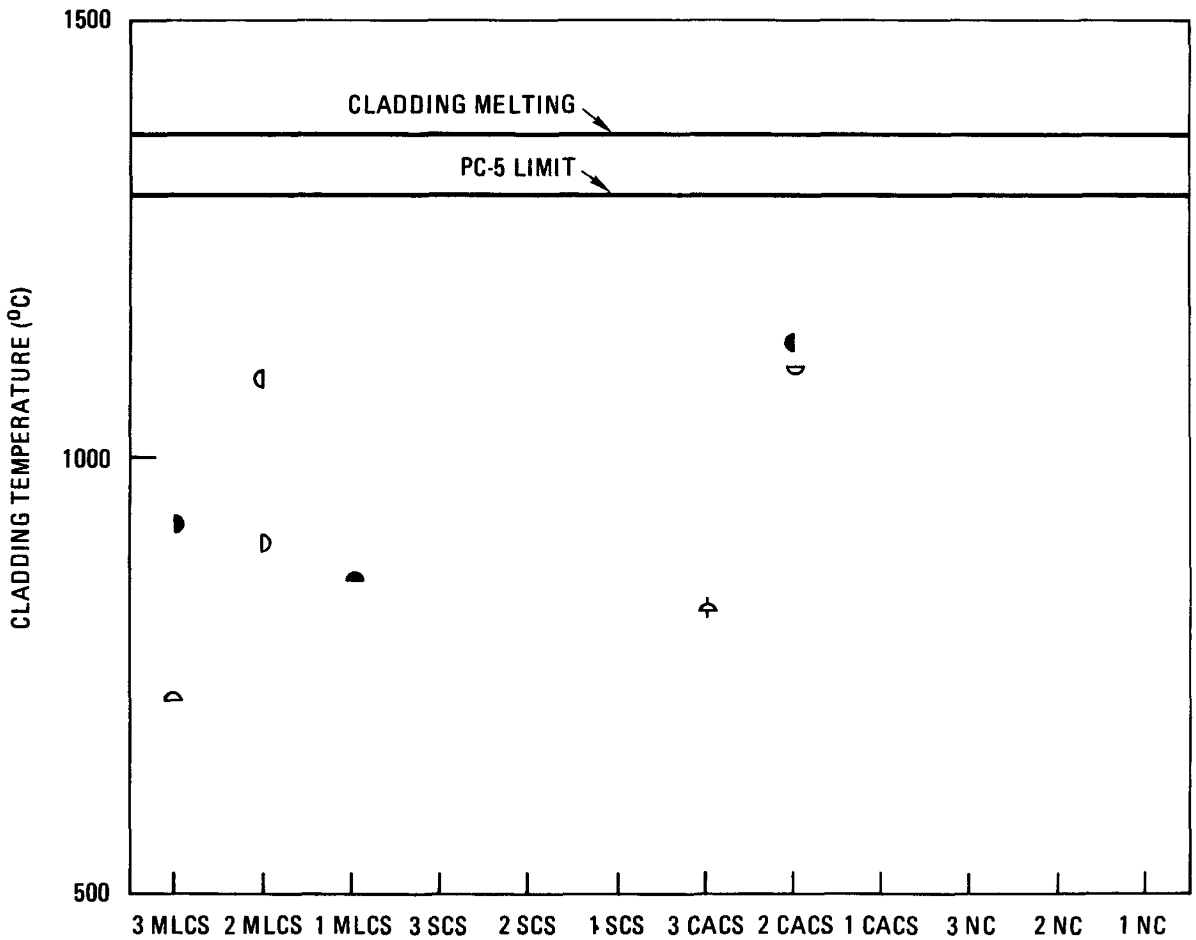

Fig. 5-28. Summary of core cooling performance in event category of decrease in reactor coolant inventory; maximum fuel cladding temperature

SYMBOL	INITIATING EVENT	ADDITIONAL FAILURES	PLANT CONDITION
	DBDA	0	PC-5
	DBDA	SINGLE	PC-5
	DBDA	DOUBLE	PC-5
	DBDA	DOUBLE	BEYOND PC-5
	DBDA WITH A LARGE LEAK AREA	SINGLE	BEYOND PC-5
	DBDA WITH LOSP	MULTIPLE	PC-5
	DBDA WITH LOSP	MULTIPLE	PC-5
	DBDA WITH LOSP AND NC IN HEAT SINK	MULTIPLE	BEYOND PC-5

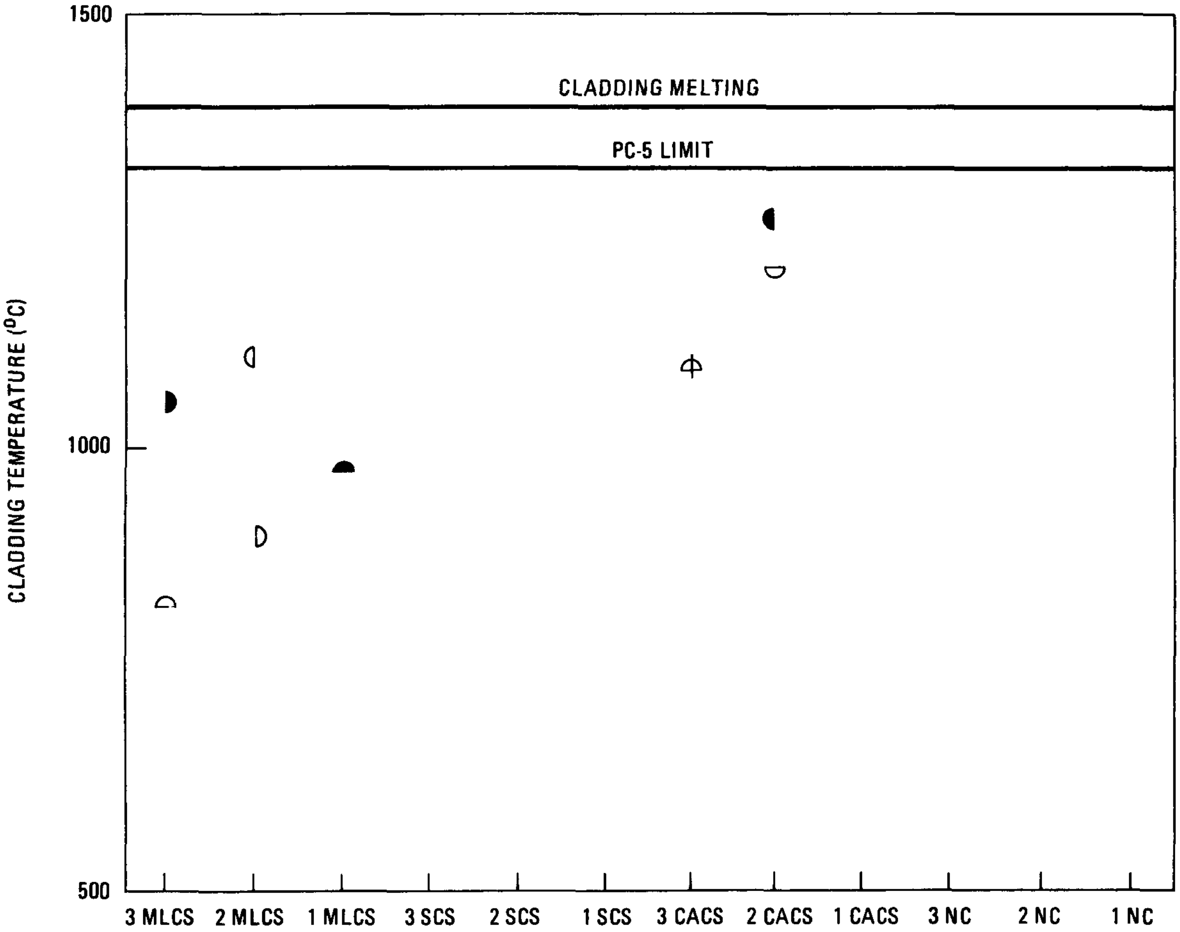

Fig. 5-29. Summary of core cooling performance in event category of decrease in reactor coolant inventory; maximum blanket cladding temperature

5.5. REACTIVITY ACCIDENTS, INADVERTENT CONTROL ROD WITHDRAWAL

An inadvertent control rod withdrawal results in increased reactor power and correspondingly increased core heat flux. Since the heat extraction from the steam generators lags the core power generation, both fuel cladding and primary coolant temperatures experience a net increase. Unless terminated by manual or automatic action, the inadvertent rod withdrawal with its high heat flux and coolant temperatures could exceed the fuel or cladding temperature design limits.

Therefore, the PPS has several automatic features which can terminate such a postulated accident:

1. High steam-generator gas-inlet temperature actuates a reactor trip if two-out-of-three channels exceed the temperature setpoint of 565°C (1050°F).

2. Power range neutron flux instrumentation actuates a reactor trip if two-out-of-three channels exceed the overpower setpoint of 110%. Neutron flux instrumentation has both primary and secondary trip inputs.

3. Exceeding the power to core flow ratio setpoint of 1.3 in two-out-of-three channels results in reactor trip.

4. Exceeding the power to feedwater flow ratio setpoint of 1.5 in two-out-of-three channels results in reactor trip.

5. Exceeding the rate of power increase limit of 30%/min in two-out-of-three channels results in a reactor trip.

Section 4.4 describes the operation of these and other reactor trips. This section discusses rod withdrawal transients initiated from full design power and 30% of rated power and briefly discusses a simultaneous failure in the SCS.

5.5.1. Inadvertent Rod Withdrawal from Full Power

5.5.1.1. Identification of Causes and Accident Description. Whether due to equipment malfunction or erroneous operator action, inadvertent rod withdrawal assumes that a single control rod assembly is withdrawn at normal shim speed until terminated by an automatic PPS feature.

As the control rod is withdrawn, reactor power rises. Since a band width is allowed about the normal operating point for the various primary and secondary system parameters, no automatic functions act immediately. Thus, power continues to rise until it reaches 110% of nominal full power. At this point, the reactor is tripped, an LOSP is assumed, and the main circulators begin coasting down. When the circulator speed falls below 30%, the SCS starts. Shutdown core cooling is then maintained by the SCS.

5.5.1.2. Analysis of Effects and Consequences. The detailed system computer program FASTRAN was used to analyze the rod withdrawal transient (Section 5.1.6.1.).

Section 5.1.3.1. discusses the plant characteristics and the initial conditions. To give conservative results, the analyses used the conservative model, which includes all the system parameter uncertainty margins (see Table 5-8).

Figure 5-30 shows the variation in major plant parameters during a rod withdrawal accident from full power. Fuel and clad temperatures are seen to rise with reactor power. However, when reactor power reaches 110%, 82 s into the transient, the PPS initiates a reactor trip. Immediately after scram, the power-to-flow ratio decreases below unity, and the rise in clad and fuel temperatures is turned around. Also, at the time of reactor trip, an LOSP is assumed. Figure 5-30(e) shows that circulator speed decreases as

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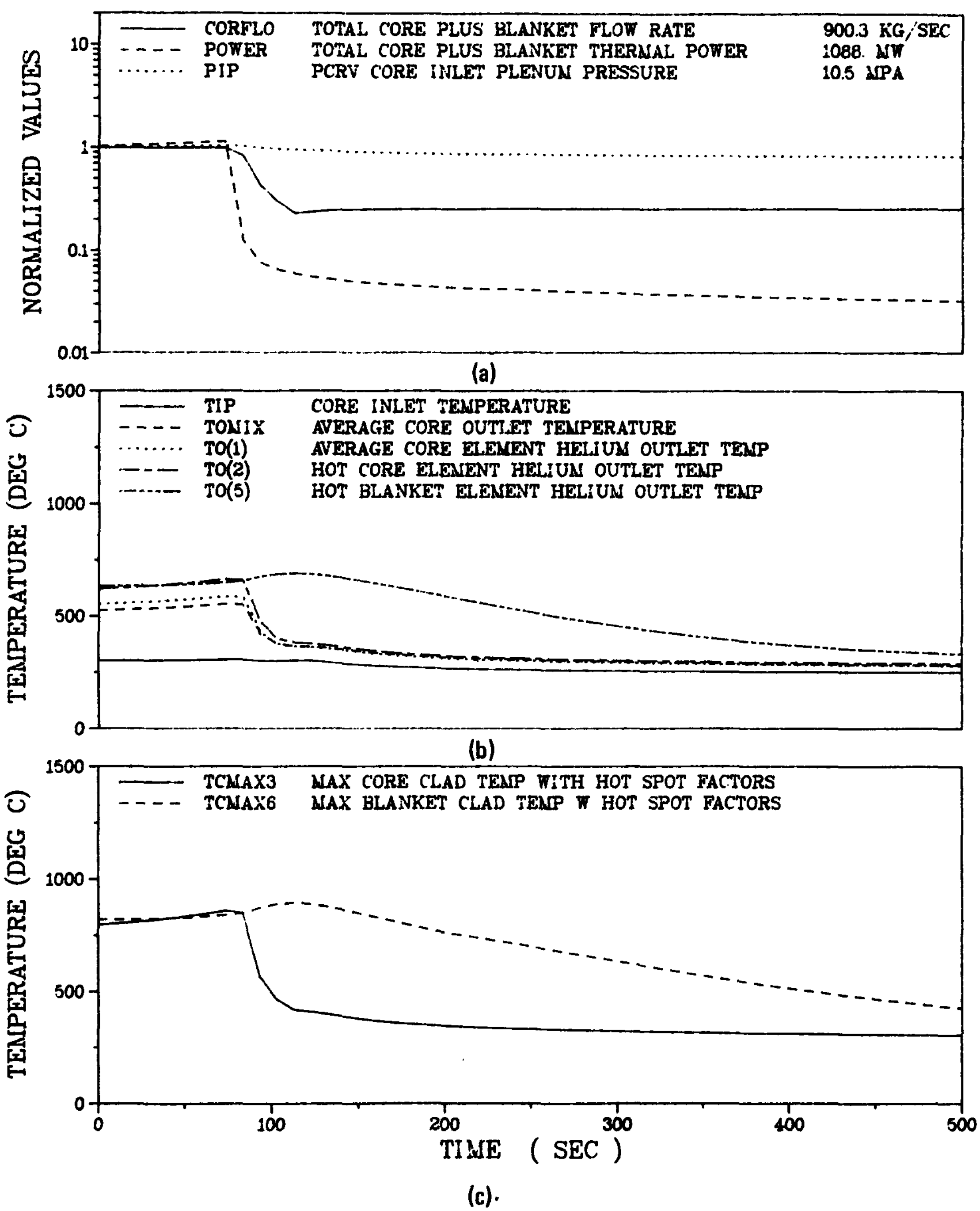


Fig. 5-30. Inadvertent control rod withdrawal at full power: (a) core power, flow, and pressure, (b) inlet and outlet temperatures, (c) maximum cladding temperatures (sheet 1 of 2)

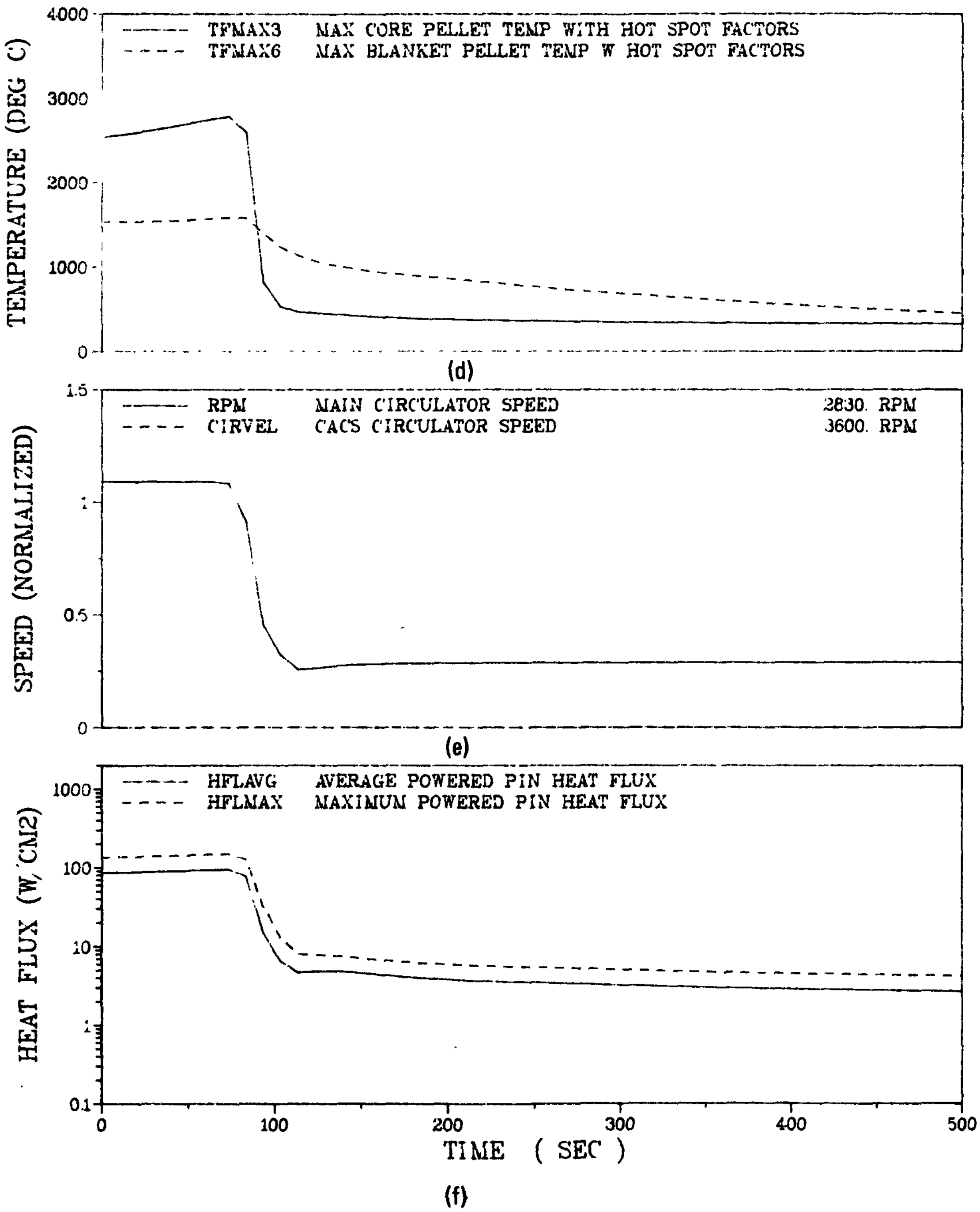


Fig. 5-30. Inadvertent control rod withdrawal at full power: (d) maximum fuel temperature, (e) circulator speeds, (f) cladding heat flux (sheet 2 of 2)

the motor coasts down until speed decreases below 30%. Then the SCS assumes core cooling.

5.5.1.3. Radiological Consequences. The radiological consequences of this event are negligible, since no fuel damage is predicted and no containment boundary is compromised.

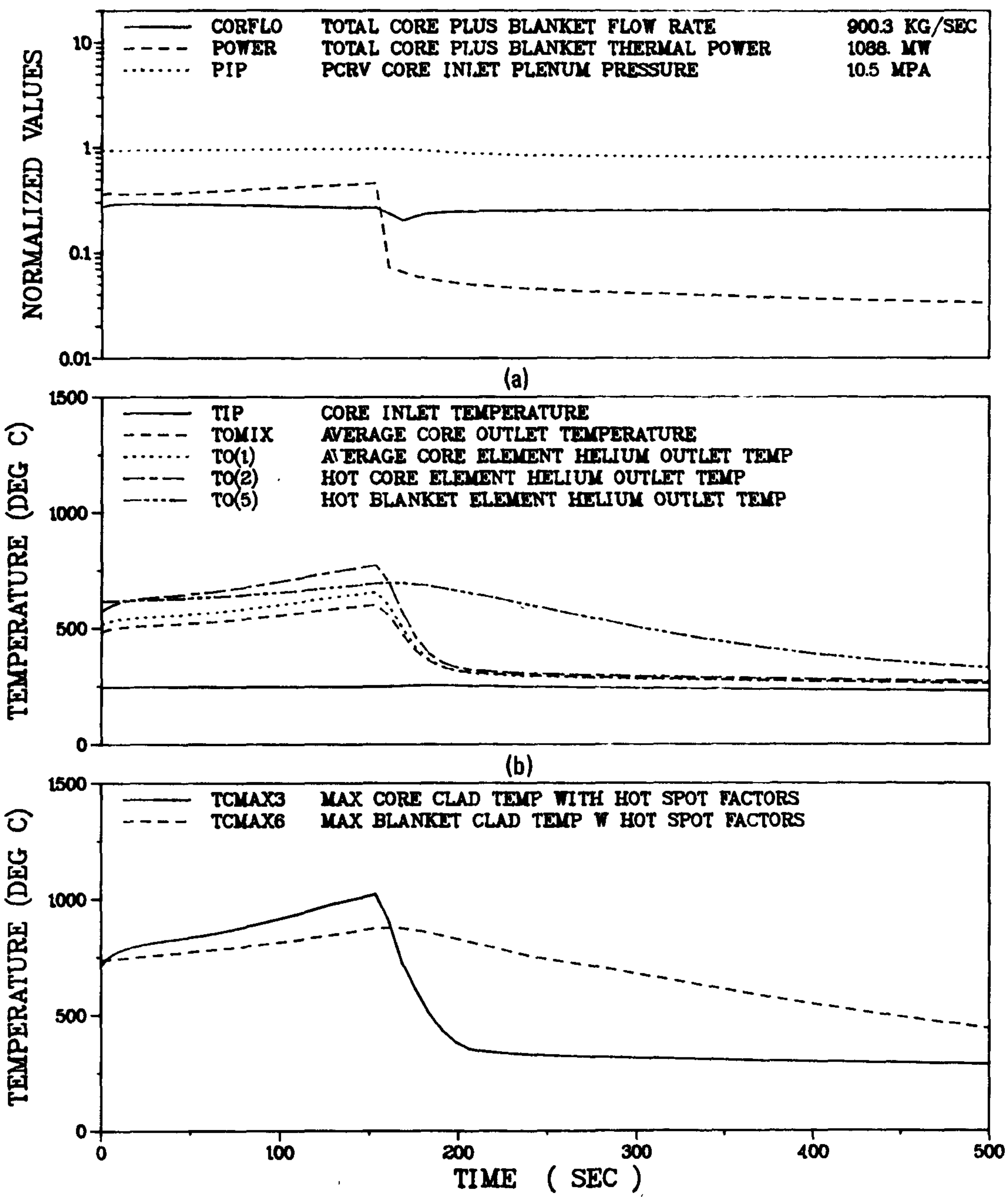
5.5.1.4. Conclusions. The analysis shows that the fuel and blanket cladding temperatures, peak centerline fuel temperatures, and the primary coolant temperatures following a rod withdrawal transient from full power are maintained below PC-3 limits.

5.5.2. Inadvertent Rod Withdrawal from 30% Power

5.5.2.1. Analysis of Effects and Consequences. Figure 5-31 shows major plant parameters during a rod withdrawal accident from 30% power. The accident beginning at 30% power differs from that at 100% power primarily in that the transient is no longer terminated by the high reactor power trip setpoint. Instead, as can be seen in Fig. 5-31(a), the reactor is tripped at 155 s by high steam generator inlet temperature. Because the time to trip is substantially longer in this case, the cladding temperatures rise markedly higher than the peak temperatures in the case of full initial power. However, because initial fuel temperatures start much lower, the peak fuel temperature is much lower than the full initial power case.

5.5.2.2. Radiological Consequences. The radiological consequences of this event are negligible, since no fuel damage is predicted and no containment boundary is compromised.

5.5.2.3. Conclusions. The analysis shows that the fuel and blanket cladding temperatures, peak centerline fuel temperatures, and the primary coolant temperatures all are maintained within PC-4 limits for a inadvertent rod withdrawal initiated from 30% power.



(c)

Fig. 5-31. Inadvertent control rod withdrawal at 30% power: (a) core power, flow, and pressure, (b) inlet and outlet temperatures, (c) maximum cladding temperatures (sheet 1 of 2)

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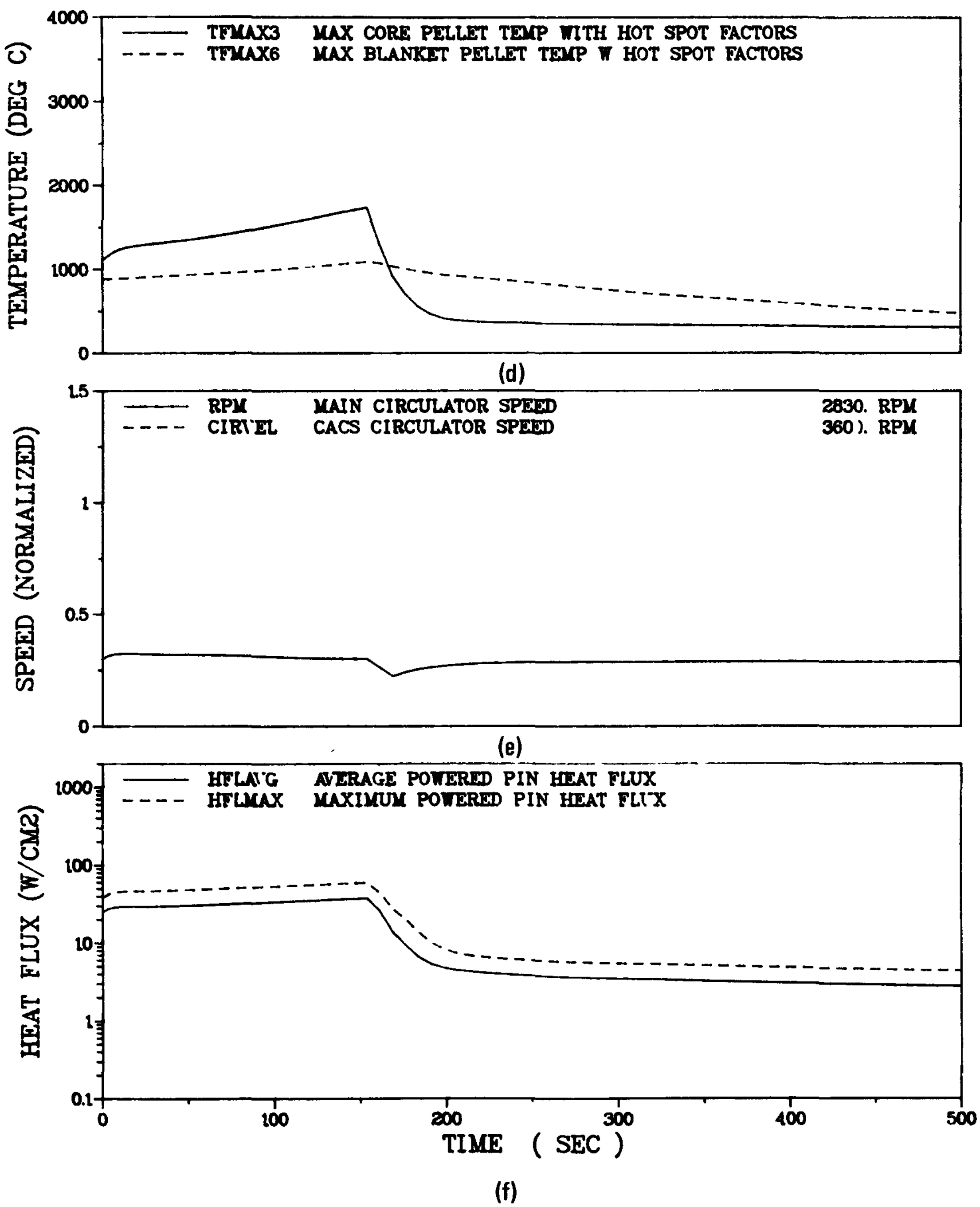

Fig. 5-31. Inadvertent control rod withdrawal at 30% power: (d) maximum fuel temperature, (e) circulator speeds, (f) cladding heat flux (sheet 2 of 2)

5.5.3. Summary and Conclusion for Category of Reactivity Accident

Sections 5.5.1 and 5.5.2 show that ample core cooling can be provided by three SCS loops following inadvertent control rod withdrawal accidents with reactor trip. The peak cladding temperature reached in these reactivity events essentially depends on the ability of the PPS to detect and respond in time, not on the RHR system cooling capacity. Analysis shows that a single failure of a SCS loop has no effect on the peak cladding temperature.

To provide a perspective for depth of protection provided by the RHR systems, Figs. 5-32 and 5-33 summarize the results of all cases of this event category, including the cases of simultaneous loss of an SCS loop. Figures 5-32 and 5-33 show maximum fuel and blanket cladding temperatures, respectively, with an abscissa indicating number and type of the RHR systems loops used. These figures indicate adequate core cooling performance and significant depth of redundancy in available backup RHR systems.

5.6. LOSS OF FORCED CIRCULATION (LOFC)

This section discusses the ability of the GCFR to respond to events resulting in a total LOFC capability. Recently, the plant tolerance to a complete loss of the station ac power for up to 2 h has been an important licensing consideration. This so-called 2 h station blackout is included in the kind of accidents that are mitigated by natural circulation RHR.

A design criteria of the GCFR is that any credible events or event sequences can be handled by one of several redundant cooling systems utilizing forced circulation. Therefore, any events requiring the use of natural circulation are considered to be beyond ANS PC-5. However, limits for these cases are considered to be the same as those for PC-5 events.

This section discusses the nominal performance of the CACS in its natural circulation mode. Additionally, while the inclusion of natural circulation as an engineered safety system is already considered to be beyond

SYMBOL	INITIATING EVENT	ADDITONAL FAILURES	MAX CLADDING TEMPERATURE (°C)	LIMIT TEMP (°C)	PLANT CONDITION
Φ	CONTROL ROD WITHDRAWAL AT FULL POWER	1	874	1100	PC-4
Ø	CONTROL ROD WITHDRAWAL AT FULL POWER	2	874	1300	PC-5
⊖	CONTROL ROD WITHDRAWAL AT 30% POWER	1	1025	1100	PC-4
⦸	CONTROL ROD WITHDRAWAL AT 30% POWER	2	1025	1300	PC-5

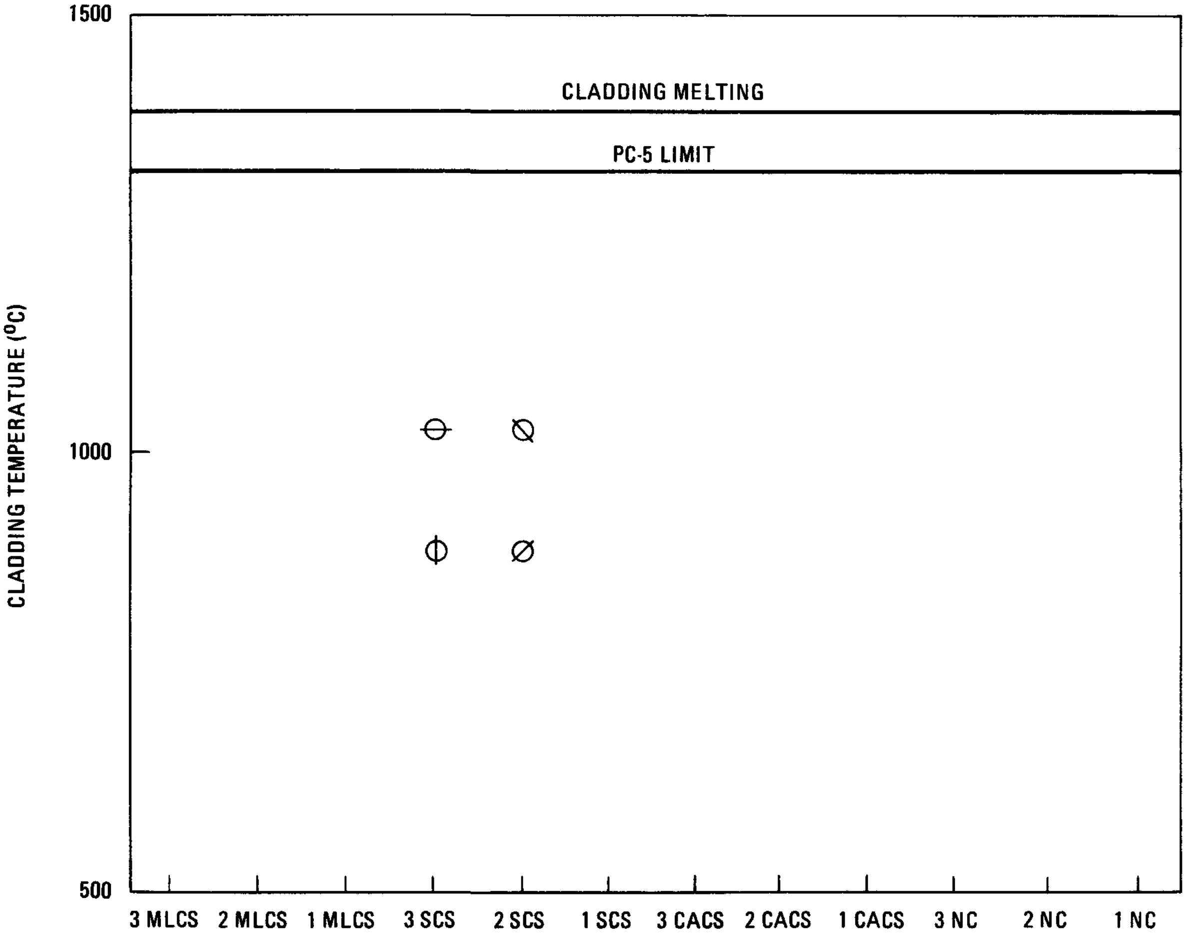

Fig. 5-32. Summary of core cooling performance in event category of reactivity accident; maximum fuel cladding temperature

SYMBOL	INITIATING EVENT	ADDITONAL FAILURES	MAX CLADDING TEMPERATURE (°C)	LIMIT TEMP (°C)	PLANT CONDITION
○	CONTROL ROD WITHDRAWAL AT FULL POWER	1	895	1100	PC-4
Ø	CONTROL ROD WITHDRAWAL AT FULL POWER	2	895	1300	PC-5
⊖	CONTROL ROD WITHDRAWAL AT 30% POWER	1	877	1100	PC-4
⦸	CONTROL ROD WITHDRAWAL AT 30% POWER	2	877	1300	PC-5

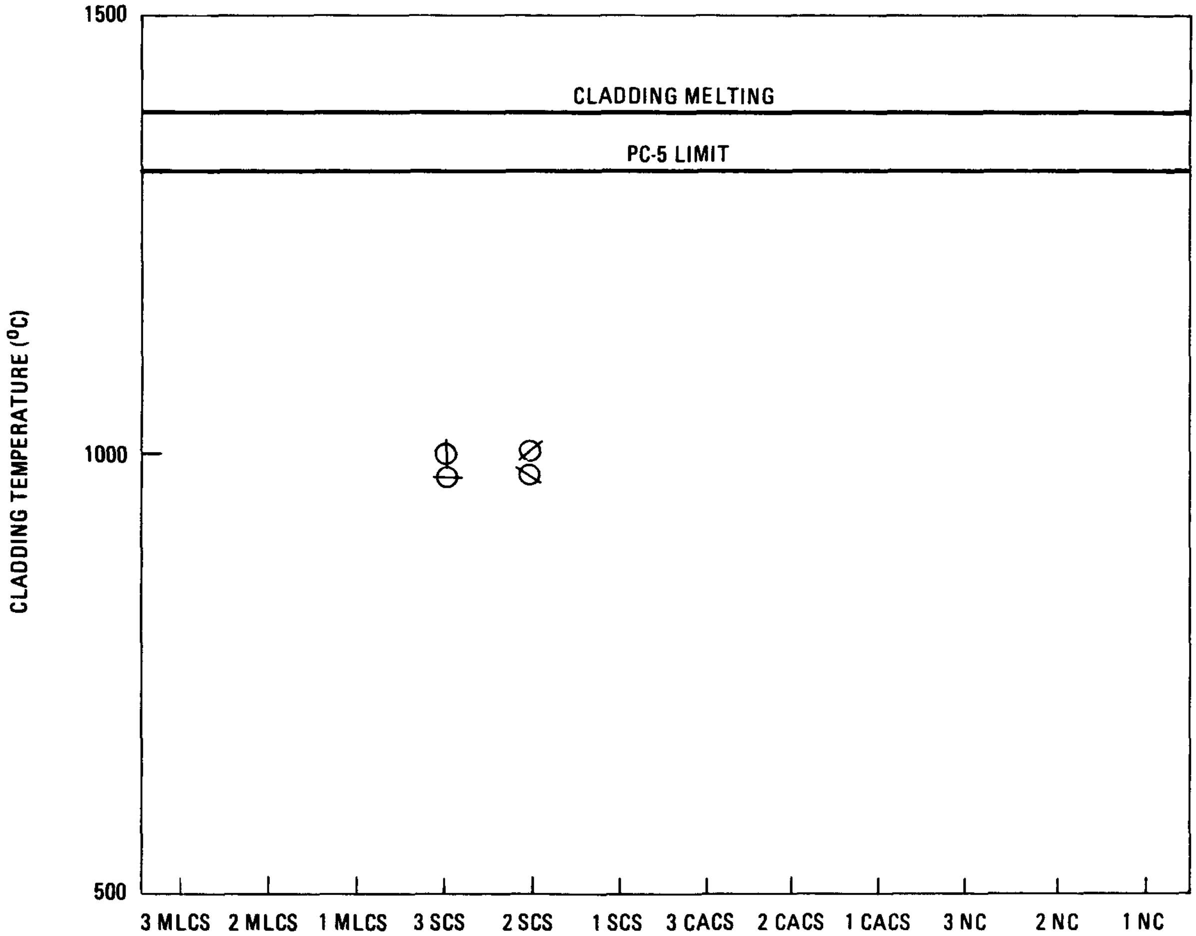

Fig. 5-33. Summary of core cooling performance in event category of reactivity accident; maximum blanket cladding temperature

the design basis and unnecessary in terms of meeting safety and reliability goals for any PC up through PC-5, to fully illustrate the depth of the GCFR core cooling capability, an additional case was analyzed assuming, further, that one CACS loop is not available for natural circulation. The natural circulation core cooling performance is evaluated for both LOFC at full pressure (Sections 5.6.1 and 5.6.2) and LOFC during refueling mitigated by primary coolant repressurization (Section 5.6.3).

5.6.1. Natural Circulation on Three Auxiliary Loops

5.6.1.1. Identification of Causes and Accident Description. A complete loss of forced reactor coolant flow presupposes a loss of the MLCS and the failure of the two independent safety-class forced circulation systems, the SCS and CACS. Such a scenario results from LOSP and loss of onsite power due to failure of the emergency generators. Current licensing requires tolerance of the station blackout for up to 2 h. The GCFR system automatically leads to natural circulation core cooling, which can continue indefinitely.

When a loss of circulation drive power occurs, the motors begin to coast down, and the reactor is scrammed. However, the inertia of the circulator motor is such that during the first 90 s after scram the power-to-flow ratio is always less than unity. At 23 s, the main circulators reach 30% speed, and the SCS pony motors are assumed to fail to start. Eighty-four seconds after the loss of circulator power (with the circulators at about 10% speed, the gravity-shutting MLIVs close against the circulator head, while the gravity-opening ALIVs open. Assuming that the auxiliary circulators also have failed to start, the CACS begins operation in its natural circulation mode.

Ultimate heat rejection proceeds from the CAHE to the atmosphere via natural circulation in the CACWS water loop and natural draft air flow through the ALC.

5.6.1.2. Analysis of Effects and Consequences. The natural circulation transients are analyzed using the detailed system computer program FASTRAN (Section 5.1.6.1.).

Section 5.1.3.1. discusses the plant characteristics and the initial conditions. To give conservative results, this analysis uses the conservative model, which allows for all the system parameter uncertainty margins (see Table 5-8). The plant control system is assumed to be in the automatic mode to delay the trip and to hold the core at full power longer. During normal plant operation with the main loops, helium leakage through the closed ALIV fills the cold leg duct of the auxiliary loops with helium at the core inlet (lower plenum) temperature, which is cooler than the core outlet (upper plenum). The helium temperature distribution thus established around the auxiliary loop is conducive to initiate natural circulation. Parasitic heat loss through the CAHE during this mode of operation maintains an elevated temperature for the CACWS hot leg pipe to support continued natural circulation if the mode changes.

In the case of transfer to the natural circulation mode following the CACS forced-circulation mode, the temperature distributions in the helium and water loops are also favorable to initiate natural circulation, since the flow paths and the directions are unchanged.

Figure 5-34 shows variation of the major plant parameters during natural circulation on the three CACS loops. Figure 5-34(a) illustrates power level, core flow, and coolant pressure during the transient. The reactor is maintained at full power following circulator trip until scrammed due to loop shutdown leading to loss of feedwater flow. Coolant flow decreases during main circulator coastdown. At 84 s, the circulator reaches 9.6% speed, and the reverse pressure drop across the ALIVs falls below a threshold value. With insufficient back pressure to continue supporting the ALIVs closed, these gravity-opening valves drop open, the MLIVs shut, and natural circulation cooling begins.

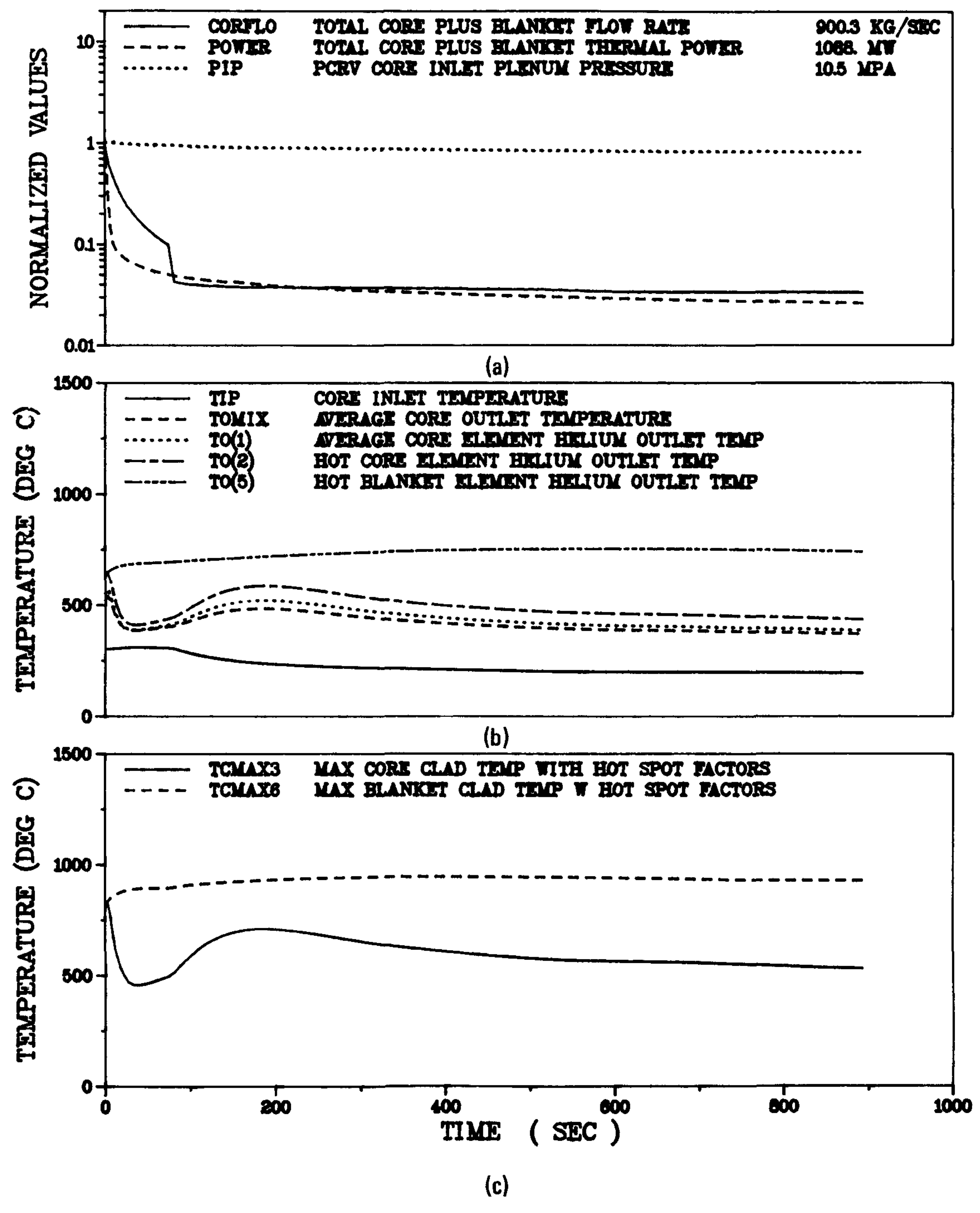

Fig. 5-34. LOFC with forced circulation, three CACS loops: (a) core power, flow, and pressure, (b) inlet and outlet temperatures, (c) maximum cladding temperatures (sheet 1 of 3)

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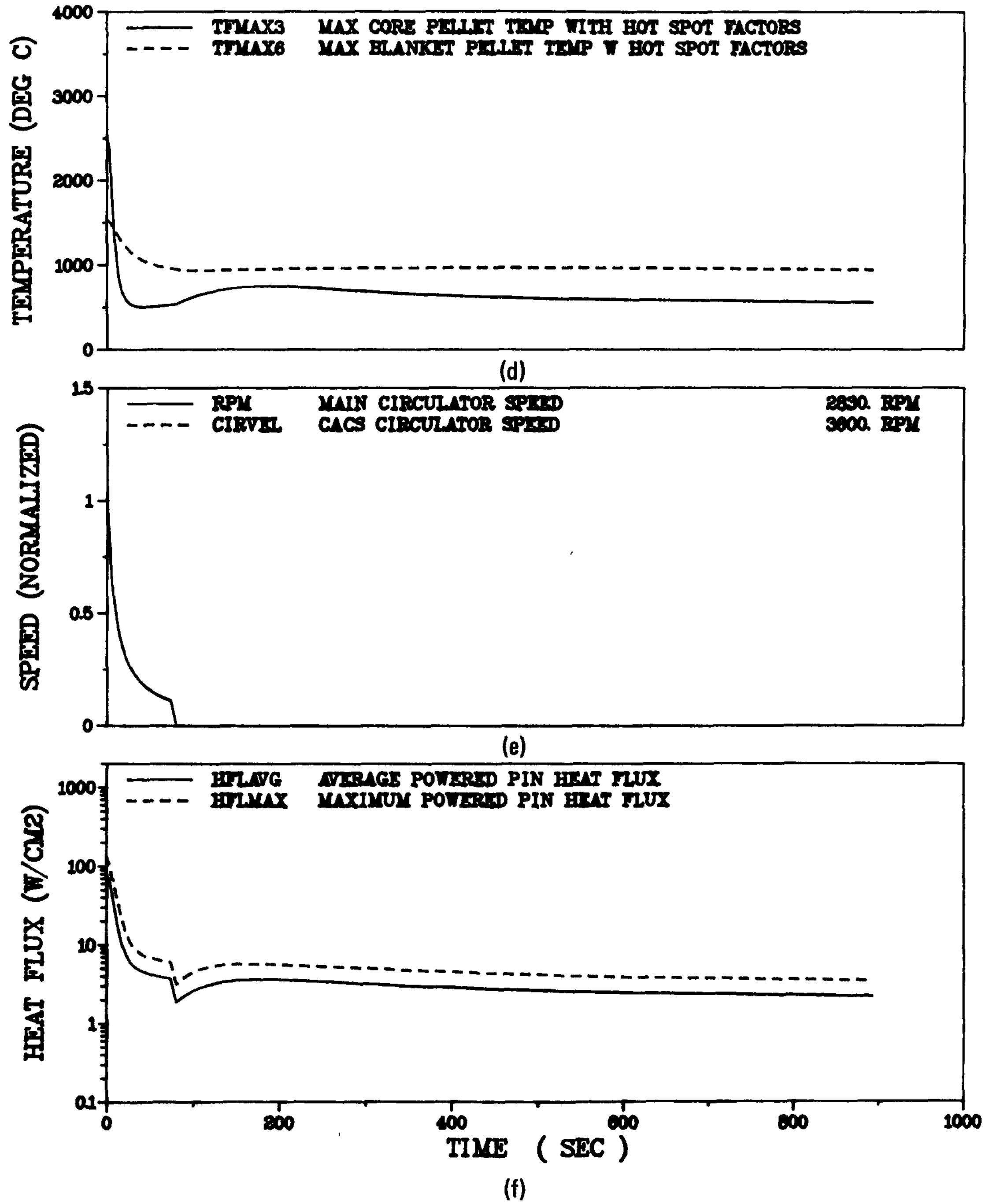

Fig. 5-34. LOFC with forced circulation, three CACS loops: (d) maximum fuel temperature, (e) circulator speed, (f) cladding heat flux (sheet 2 of 3)

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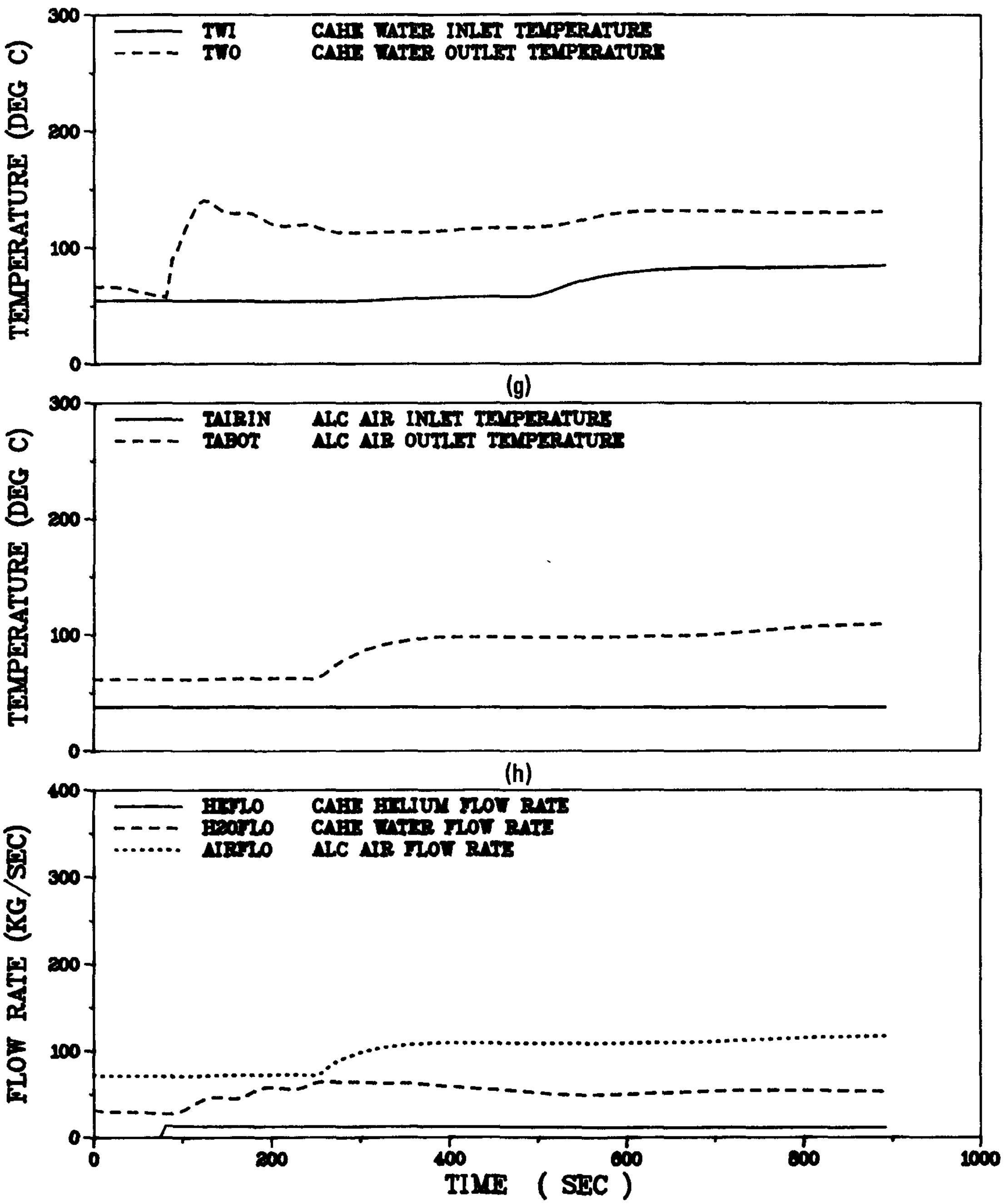

(i)

Fig. 5-34. LOFC with forced circulation, three CACS loops: (g) CAHE fluid temperatures, (h) ALC fluid temperatures, (i) CAHE and ALC flow rates (sheet 3 of 3)

Figure 5-34(b) shows core inlet and outlet gas temperatures. The decrease in inlet temperatures after shifting to the auxiliary loops is due to the low CAHE outlet temperature relative to that of the steam generators. This lower inlet temperature explains the very low hot spot cladding temperatures in the fuel and blanket, as seen in Fig. 5-34(c). Both rods reach maximum temperatures substantially below the PC-5 limits (see Section 5.1.2). Figure 5-34(e) shows coastdown and termination of the main circulator speed, leading to LOFC. Figure 5-34(g,h,i) shows CACWS system transient response. The air and water flow rates seen before 84 s are due to the aforementioned parasitic heat losses in the auxiliary loops during main loop operation. After transfer to the auxiliary loops, the hot leg water temperature rises, causing the flow to increase. After a delay due to the water transit time, air flow also increases. Water temperatures are maintained well below the system saturation temperature of the system of 302°C (577°F).

5.6.1.3. Radiological Consequences. The radiological consequences of this event are negligible, since no fuel damage is predicted and no containment is boundary compromised.

5.6.1.4. Conclusions. The analysis shows that the fuel and blanket cladding and the primary coolant temperatures following a complete loss of forced circulation with station blackout are maintained below PC-5 limits with substantial margins. The CACWS water temperature also maintains adequate margin to the 302°C (577°F) boiling point throughout the transient.

5.6.2. Natural Circulation RHR on Two CACS Loops

The need for natural circulation RHR is designed to be beyond the limits of any deterministic criteria. Nevertheless, to better illustrate the full depth of the GCFR core cooling capability, an additional case was analyzed assuming loss of a single CACS loop.

Several scenarios can be hypothesized in which a CACS loop is rendered inoperable for natural circulation. These include failure of the ALIVs to open or impaired heat removal capability of the CACWS. By far, the most severe of these is a valve failure in which no credit is taken for the heat capacity of the CAHE in the affected loop. Analysis has shown that even with a single CACS loop failure, natural circulation on two auxiliary loops can always provide sufficient cooling. The results of this case are discussed below.

If, after circulator coastdown, one ALIV fails to open, only two CACS remain available for natural circulation. Figure 5-35(a) shows power level and coolant flow throughout the transient. While not as high as the flow available from three loops, reactor coolant flow is more than adequate to maintain both fuel and blanket cladding temperatures well below PC-5 limits [see Fig. 5-35(c)]. CACWS water temperatures remain well below the saturation temperature of 302°C (577°F), as seen in Fig. 5-35(g).

Radiological Consequences. The radiological consequences of both these events are negligible, since no fuel damage occurs and the primary coolant boundary is not compromised.

5.6.3. Refueling Accident, Natural Circulation on Two CACS with Repressurization

Refueling operation is conducted under depressurized conditions using slightly subatmospheric helium. Since refueling operation starts after 48 h shutdown, reactor residual heat is less than 0.5%. Core cooling during refueling is provided by one MLCS loop, one CACS loop, or the necessary number of SCS, depending on the decay heat level (Section 4.3.2.2).

In addition to these redundant forced circulation systems, the GCFR is equipped with natural circulation backup core cooling by means of emergency repressurization of the primary coolant (Section 4.5.4.2). Normal refueling would continue only when one or more backup forced-circulation capability is available. When one of the two last forced-circulation systems fails and

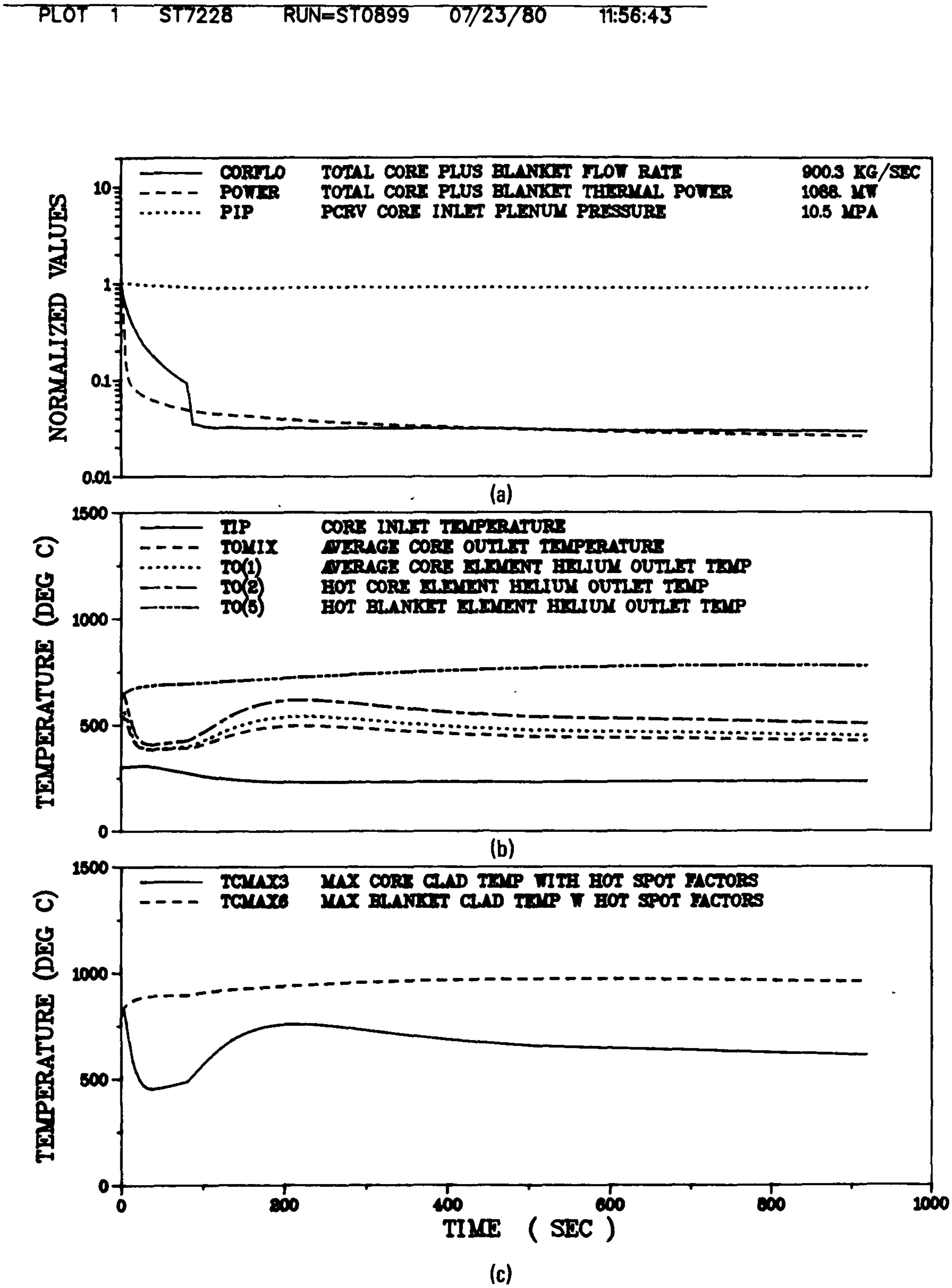

Fig. 5-35. LOFC with two CACS loops: (a) core power, flow, and pressure, (b) inlet and outlet temperatures, (c) maximum cladding temperatures (sheet 1 of 3)

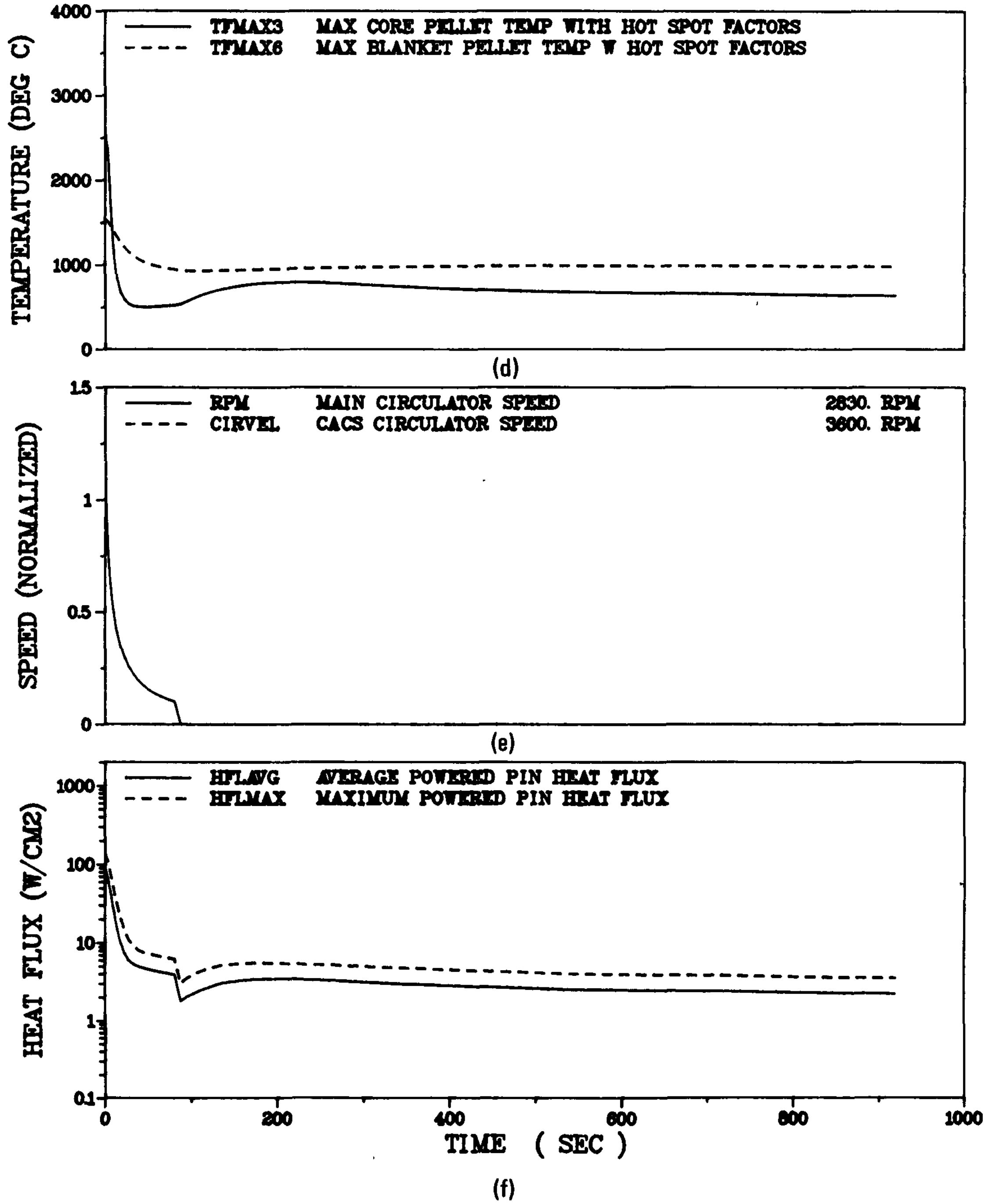

Fig. 5-35. LOFC with two CACS loops: (d) maximum fuel temperature, (e) circulator speeds, (f) cladding heat flux (sheet 2 of 3)

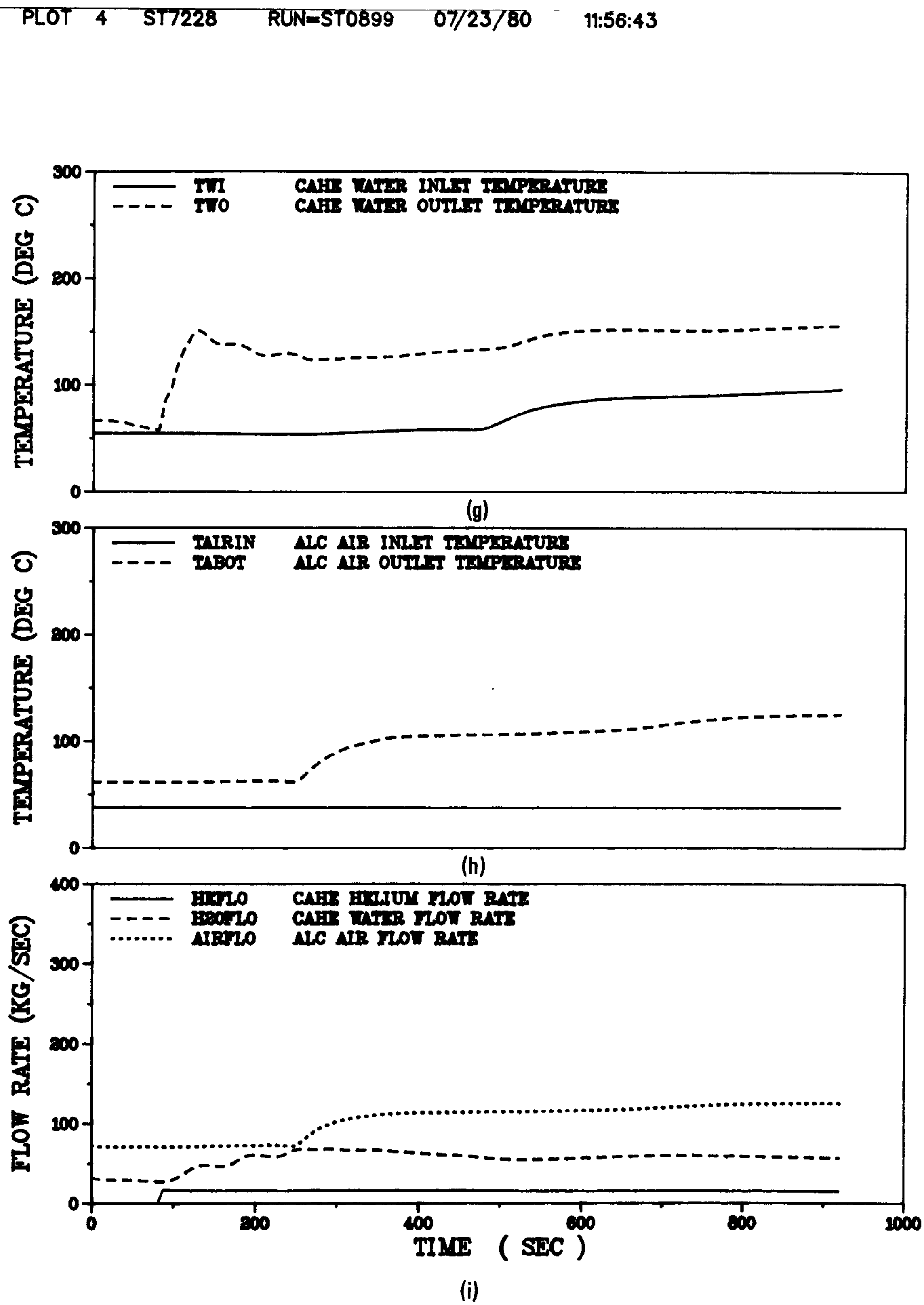

Fig. 5-35. LOFC with two CACS loops: (g) CAHE fluid temperatures, (h) ALC fluid temperatures, (i) CAHE and ALC flow rates (sheet 3 of 3)

the last one is phased in, refueling will be interrupted, and emergency repressurization will be initiated by closing the reactor isolation valve and opening the valves from the helium storage system (see Fig. 4-28). When helium is repressurized adequately, the natural circulation core cooling is established through the CACS heat transfer train similarly to that described in the previous section.

The most conservative repressurization transient scenario is that the last forced circulation system failure before repressurization is completed, even though this is highly unlikely. A case of repressurization transient starting from complete LOFC during refueling is analyzed in the following subsection.

5.6.3.1. Identification of Causes and Accident Description. Use of the CACS for core cooling during refueling presumes that the MLCS and SCS are not available, or that these systems are not used intentionally during refueling. The particular accident under consideration here is an LOFC in the auxiliary loops, which could be caused by an LOSP followed by or coincident with failure of the independent three safety class 1E power supplies. Since the MLCS and SCS are out of service, no other forced cooling capability is available. The auxiliary circulator coasts down and natural convection in the CACS loops begins. At atmospheric pressure, however, natural convection is insufficient to provide adequate core cooling and keep cladding temperatures below acceptable limits. Thus, emergency repressurization of the PCRV from the helium storage tanks is initiated (see Section 4.5.4.2.).

5.6.3.2. Analysis of Effects and Consequences.

Method of Analysis. The systems computer program FASTRAN was used to simulate the LOFC followed by emergency repressurization. Table 5-14 gives initial conditions for refueling with cooling on two CACS loops. These conditions assume a conservative plant model.

TABLE 5-14
INITIAL CONDITIONS, REFUELING ACCIDENT, COOLING ON 2/3 AUXILIARY LOOPS
CONSERVATIVE PLANT MODEL

Coolant inventory [kg (lb)]	281.2 (620.0)[(a)]
Power (MW)	6.01[(b)]
CAHE water flow [kg/s (lb/s)]	252.0 (555.56)[(c)]
Cooling tower air flow [kg/s (lb/s)]	134.9 (297.5)[(c)]
Cooling tower air inlet temperature [°C (°F)]	37.8 (100.0)[(c)]
Circulator speed (rpm)	3600[(d)]
Helium flow per loop [kg/s (lb/s)]	5.35 (11.79)
Core pressure drop [kPa (psi)]	2.30 (0.334)
Core Reynolds number	1312
Inlet plenum pressure [MPa (psia)]	0.10 (14.3)
High power, hot spot max core clad temperature [°C (°F)]	330 (626)
High power, hot spot max blanket clad temperature [°C (°F)]	651 (1204)
Inlet plenum temperature [°C (°F)]	106 (223)
Outlet plenum temperature [°C (°F)]	214 (417)
CAHE water inlet temperature [°C (°F)]	62.8 (145)
CAHE water outlet temperature [°C (°F)]	65.6 (150)
Cooling tower air outlet temperature [°C (°F)]	61.7 (143)

(a) Coolant inventory chosen to provide atmospheric pressure.
(b) Two days after scram, plus 10% margin.
(c) Standard CACS data, forced water and air flow.
(d) Maximum circulator speed permitted at this pressure.

The LOFC is simulated by setting the auxiliary circulator motor torque to 0 and setting the mode of heat removal in the program to total natural circulation at time 0. This implies that the primary coolant flow, the water flow in the CAHE, and the air flow in the cooling tower are all induced by natural circulation.

A 15 min delay is assumed before commencing emergency repressurization. This allows time to isolate the PCRV and accounts for repressurization having to be activated by operator command. Hence, some unavoidable delay will occur in initiating repressurization after LOFC.

When repressurization begins, the helium flow from the 103 MPa (1500 psia) storage tanks into the PCRV is choked, with a flow rate of 21.5 kg/s (47 lb/s). Repressurization is halted when 3670 kg (8092 lb) of helium have been transferred (~150 s after starting).

Results. Figure 5-36 shows the system response to an LOFC during refueling followed by emergency repressurization. Figure 5-36(a) shows the initial decrease in core coolant flow as forced circulation is lost and the subsequent rapid increase in pressure and flow when repressurization begins. The PCRV pressure attains 1.72 MPa (250 psia). The sudden drop in flow some minutes after initiation of repressurization is caused by the auxiliary circulator speed dropping to less than 5%, the minimum design speed. From this point, the circulator speed is conservatively assumed to be 0.

Figure 5-36 (b) shows core outlet and inlet temperatures. Figure 5-36(c) shows the cladding temperature in a high-power, hot-spot channel in the core (TCMAX3) and blanket (TCMAX6). TCMAX3 initially increases as forced primary coolant flow is lost, peaks at 1032°C (1890°F) after repressurization begins, decreases to 722°C (1331°F), then begins increasing again as the auxiliary circulator speed goes to 0 and core flow suddenly drops. TCMAX3 then levels out at 778°C (1433°F). The peak value of TCMAX3 is well below the PC-5 limit of 1260°C (2300°F). The slow response of TCMAX6 reflects the large thermal inertia of the blanket. This temperature

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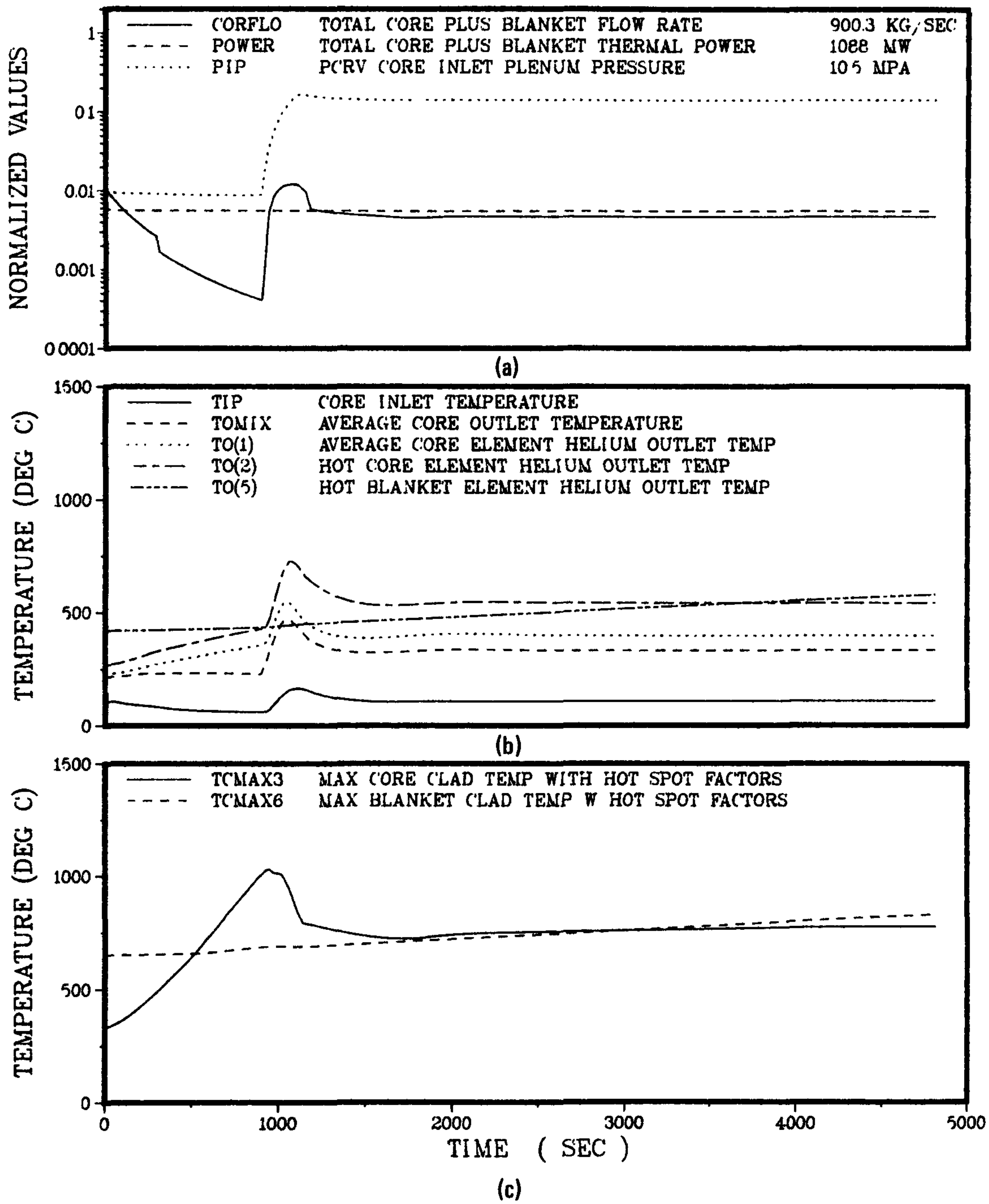

Fig. 5-36. LOFC during refueling; natural circulation with repressurization, two CACS loops: (a) core power, flow, and pressure, (b) inlet and outlet temperatures, (c) maximum cladding temperatures (sheet 1 of 4)

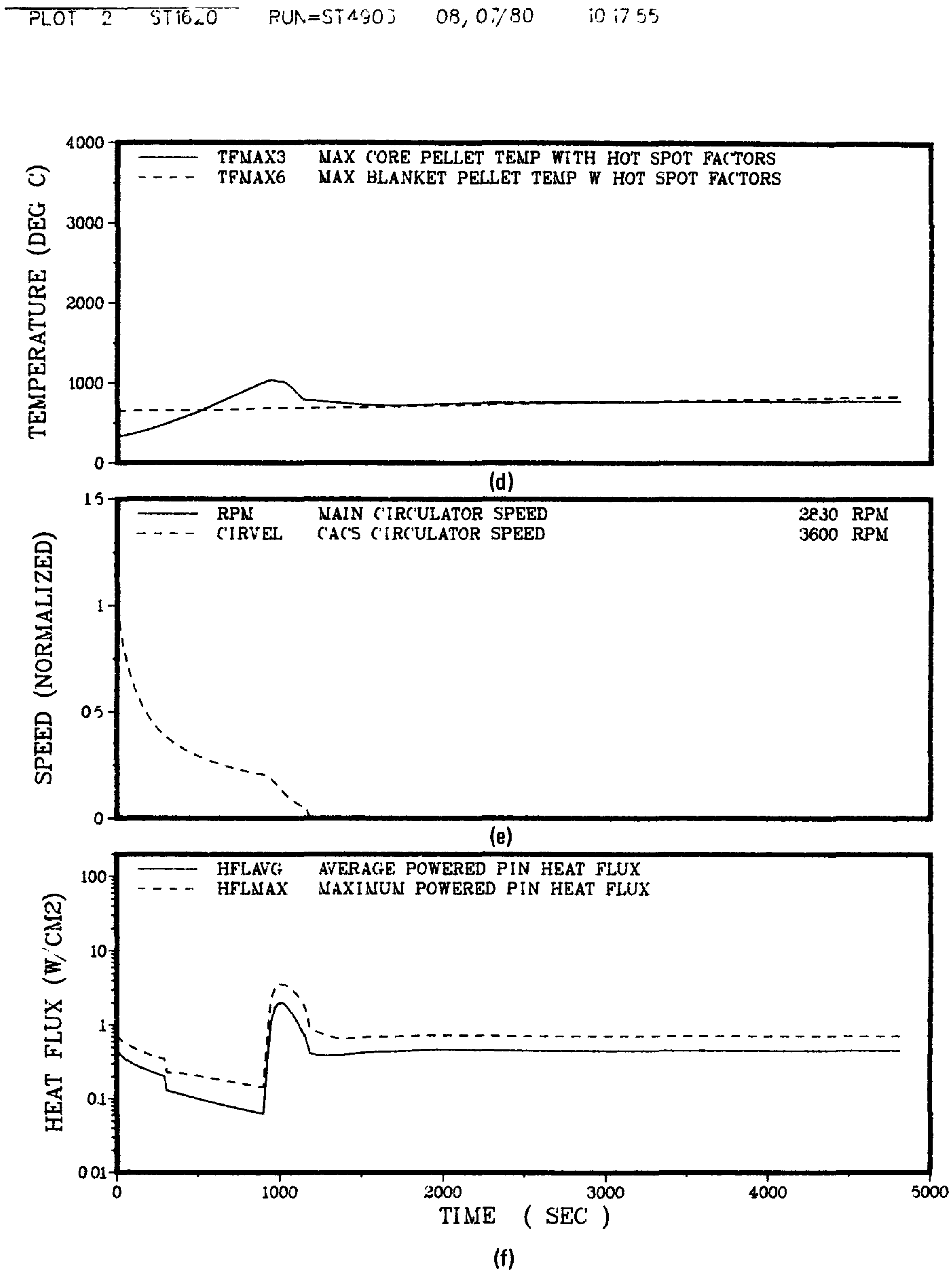

Fig. 5-36. LOFC during refueling; natural circulation with repressurization, two CACS loops: (d) maximum fuel temperature, (e) circulator speed, (f) cladding heat flux (sheet 2 of 4)

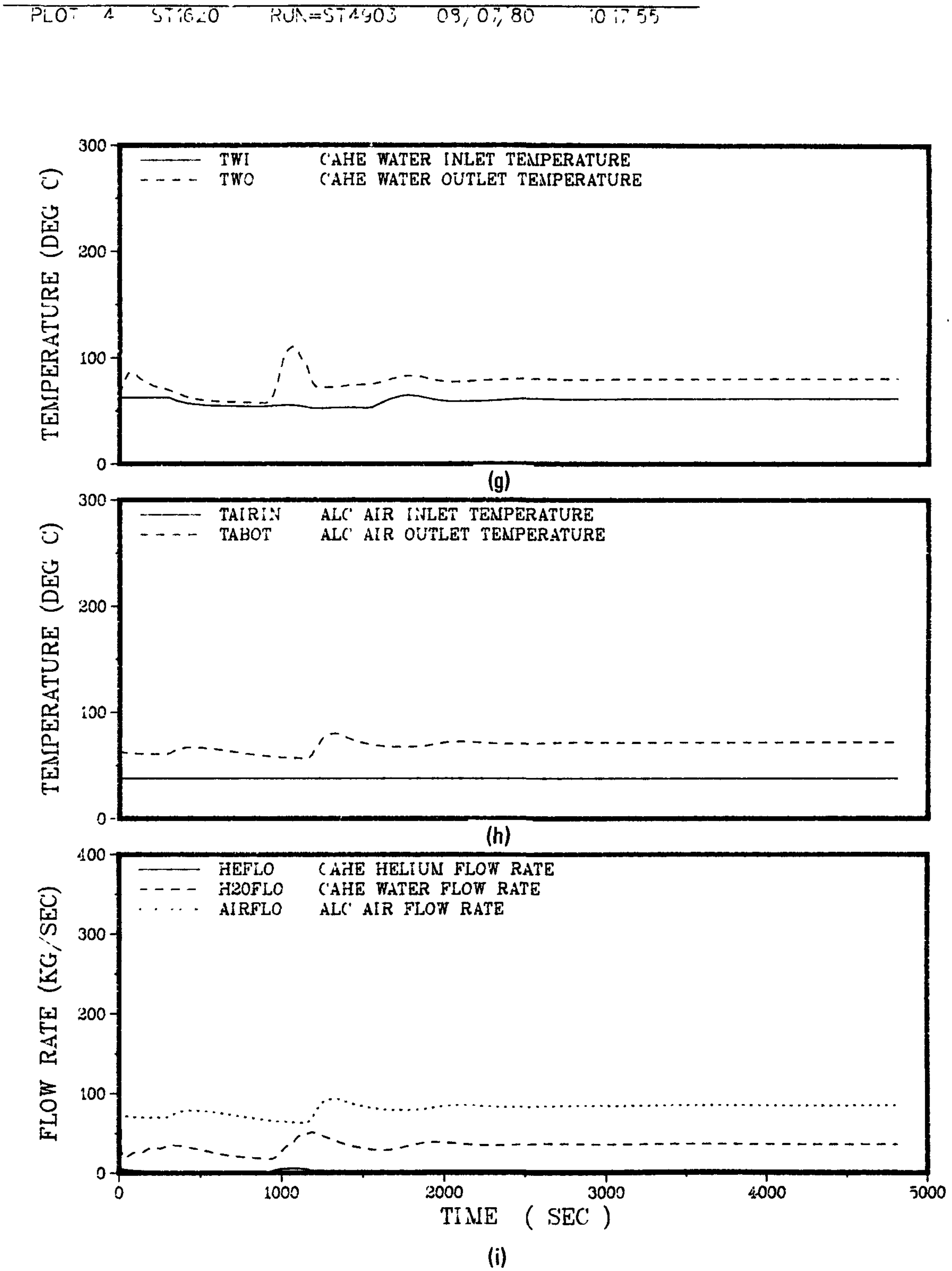

Fig. 5-36. LOFC during refueling; natural circulation with repressurization, two CACS loops: (g) CAHE fluid temperatures, (h) ALC fluid temperatures, (i) CAHE and ALC flow rates (sheet 3 of 4)

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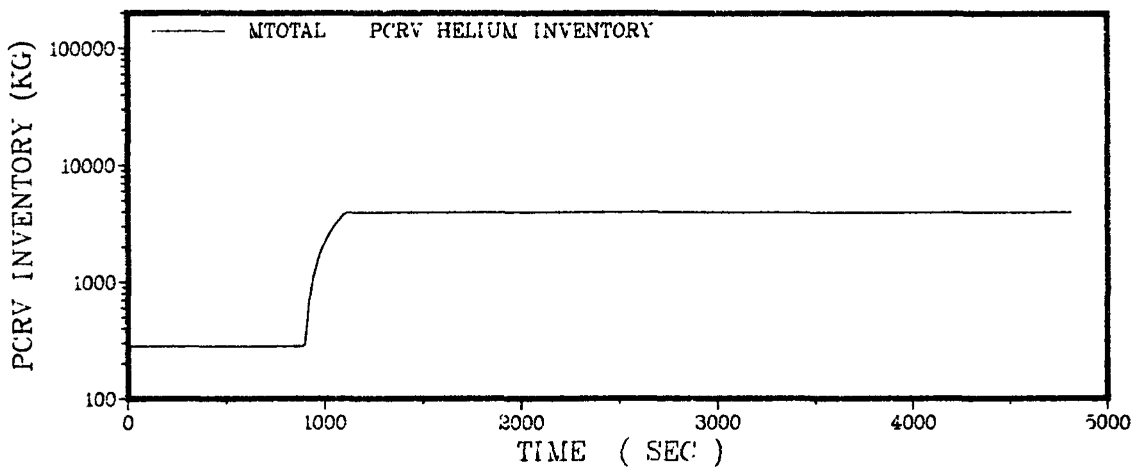

(j)

Fig. 5-36. LOFC during refueling; natural circulation with repressurization, two CACS loops: (j) primary coolant inventory (sheet 4 of 4) eventually attains a maximum of 1204°C (2200°F), which is below the PC-5 limit.

Figure 5-36(d) shows the fuel temperature for a high-power, hot-spot channel in the core (TFMAX3). The curve is similar to that of TCMAX3 [Fig. 5-36(c)]. Figure 5-36(e) shows the auxiliary circulator rotative speed (CIRVEL) coasting down.

Figure 5-36(g,h,i) illustrates CACS system parameters during the transient. Figure 5-36(g) shows the CAHE water inlet (TWI) and outlet (TWO) temperatures. The water outlet temperature increases as repressurization begins due to the increased helium flow and heat transfer in the CAHE. The water outlet temperature also increases, but the effect is delayed because of the large water loop transit time (~700 s), and the effect is damped because of the heat capacity of the CAHE pipes. The water temperature is well below 302°C (577°F) boiling point of 9 MPa (1300 psia) water.

Figure 5-36(h) shows the inlet (TAIRIN) and outlet (TABOT) temperatures in the cooling tower. Figure 5-36(i) gives flows in the CACS: the primary coolant flow per loop (HEFLO), the CAHE water flow (H2OFLO), and the cooling tower air flow (AIRFLO).

Figure 5-36(j) illustrates the total reactor helium inventory during the transient. Approximately 3670 kg (8092 lb) are transferred from the helium storage system during repressurization.

5.6.3.3. Radiological Consequences. No radiological consequences result from this event, since no primary or secondary coolant is released.

5.6.3.4. Conclusions. This analysis shows that emergency repressurization of the PCRV from the helium storage system can mitigate the effects of a complete LOFC during refueling. With a conservative plant model and two CACS loops operating, a 15 min delay is permitted before repressurizing and transferring 3670 kg (8092 lb) of additional helium to the PCRV. Cladding temperatures remain below the PC-5 limit throughout the transient.

5.6.4. Summary and Conclusions for Category of LOFC

Figures 5-37 and 5-38 summarize the maximum fuel and the blanket temperatures, respectively, obtained for the various natural circulation cases in the previous sections. Significant margins to the faulted cladding temperature limit (i.e., PC-5) are indicated for all the cases. Thus, adequate and redundant RHR is assumed to be achievable using the inherently passive and diverse system of natural coolant circulation from the core to the ultimate atmospheric heat sink.

5.7. DISCUSSION AND CONCLUSION FOR CORE COOLING PERFORMANCE

The previous sections used transient analysis to examine the GCFR system response to accidents of various categories and ANS plant conditions. Adequacy of the RHR system capability will be addressed below in two aspects: (1) performance margin with respect to meeting the design temperature limits and (2) redundancy margin with respect to number of available backup RHR systems in excess of the deterministic safety requirements (see Section 5.1.1.).

5.7.1. RHR Performance Margin

The previous sections used a conservative model to analyze the GCFR RHR performance. The conservative analysis model is defined by applying the system parameter uncertainties (Table 5-8) simultaneously in a direction most detrimental to core cooling.

To provide greater insight into the performance margin incorporated in the design, the results of the conservative model with the uncertainty margins are compared with that of the best estimate model without the uncertainty margins. For this comparison, a DBDA transient is chosen as the most limiting case among the RHR cases analyzed. Figure 5-39 shows the maximum cladding temperatures obtained by both the conservative and the best estimate models. The peak temperature difference between the two curves is

SYMBOL	INITIATING EVENT	ADDITONAL FAILURES	MAX CLADDING TEMPERATURE (^{o}C)	LIMIT TEMP (^{o}C)	PLANT CONDITION
	LOSS OF ALL FORCED CIRCULATION AT POWER	0	837	1300	BEYOND PC-5
	LOSS OF ALL FORCED CIRCULATION AT POWER	1	837	1300	BEYOND PC-5
	LOFC DURING REFUELING WITH REPRESSURIZATION	1	1032	1300	BEYOND PC-5

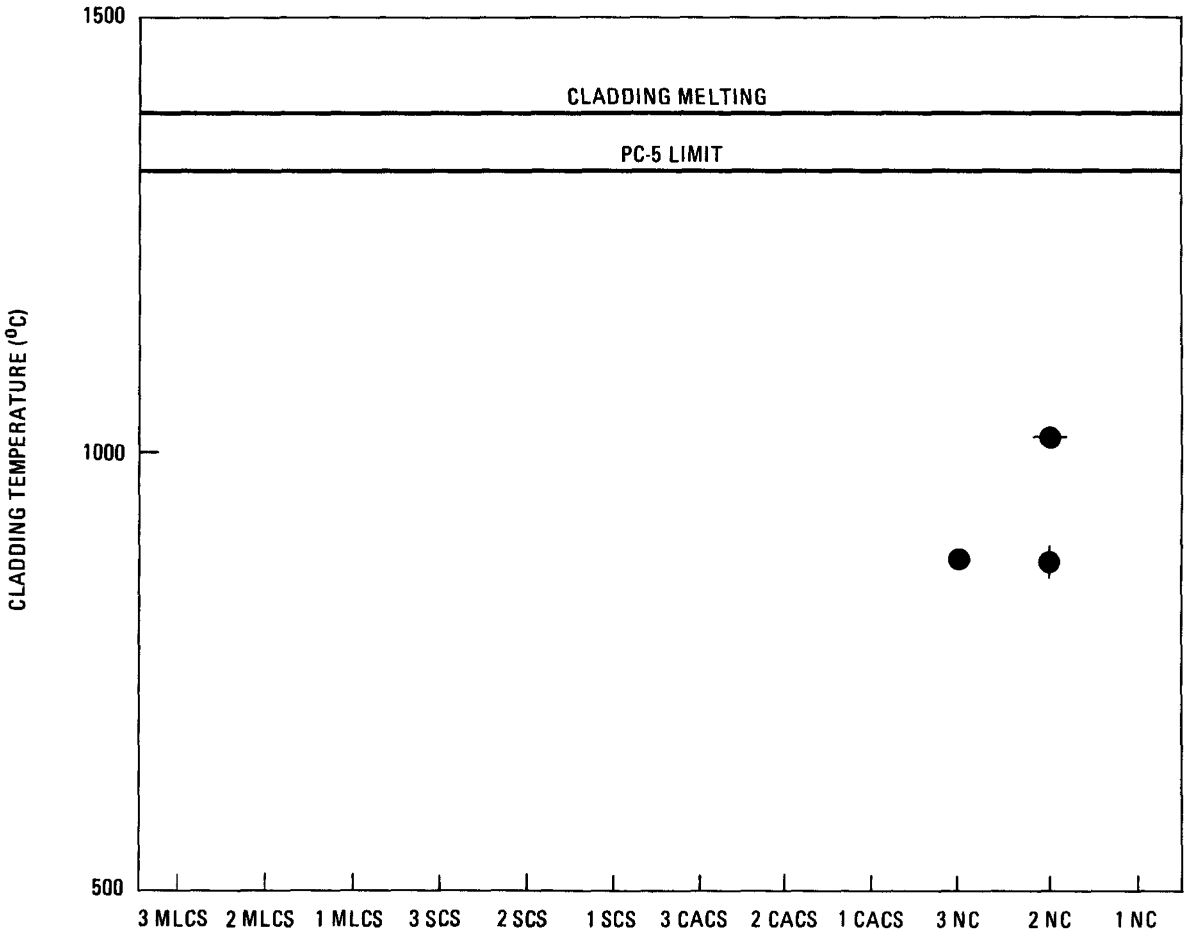

Fig. 5-37. Summary of core cooling performance in event category of LOFC; maximum fuel cladding temperature

SYMBOL	INITIATING EVENT	ADDITONAL FAILURES	MAX CLADDING TEMPERATURE (°C)	LIMIT TEMP (°C)	PLANT CONDITION
●	LOSS OF ALL FORCED CIRCULATION AT POWER	0	945	1300	BEYOND PC-5
⧫	LOSS OF ALL FORCED CIRCULATION AT POWER	1	972	1300	BEYOND PC-5
-●-	LOFC DURING REFUELING AND REPRESSURIZATION	1	1212	1300	BEYOND PC-5

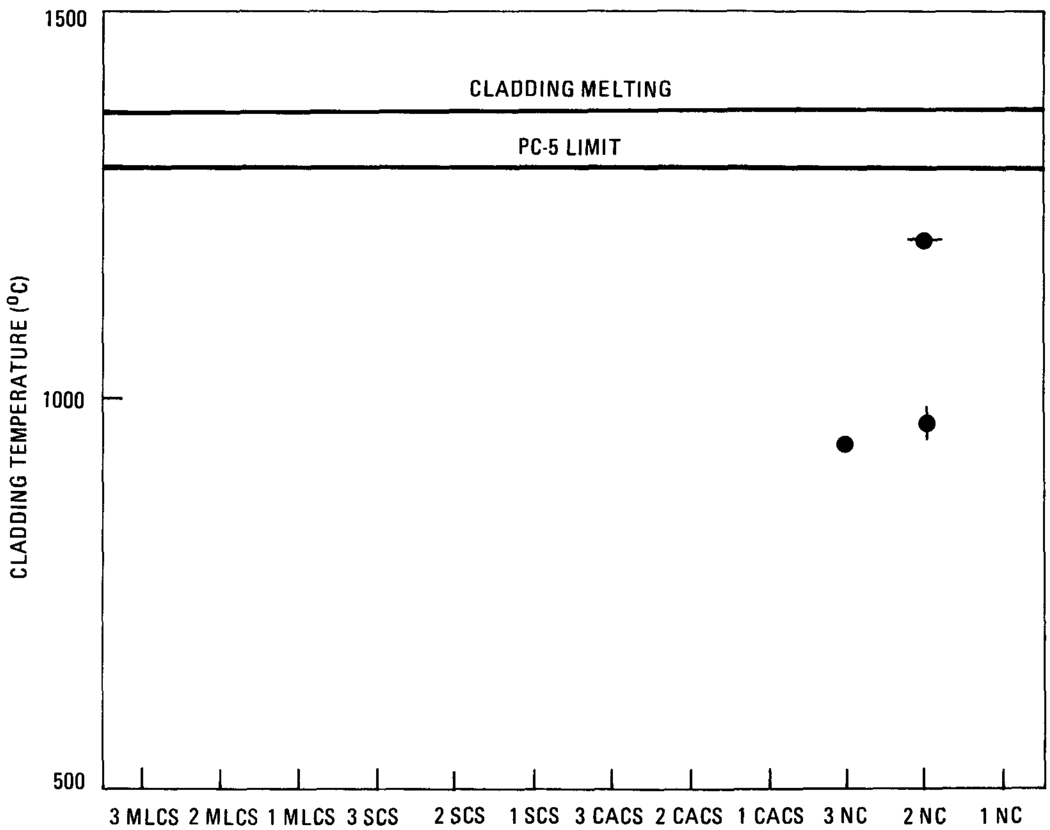

Fig. 5-38. Summary of core cooling performance in event category of LOFC; maximum blanket cladding temperature

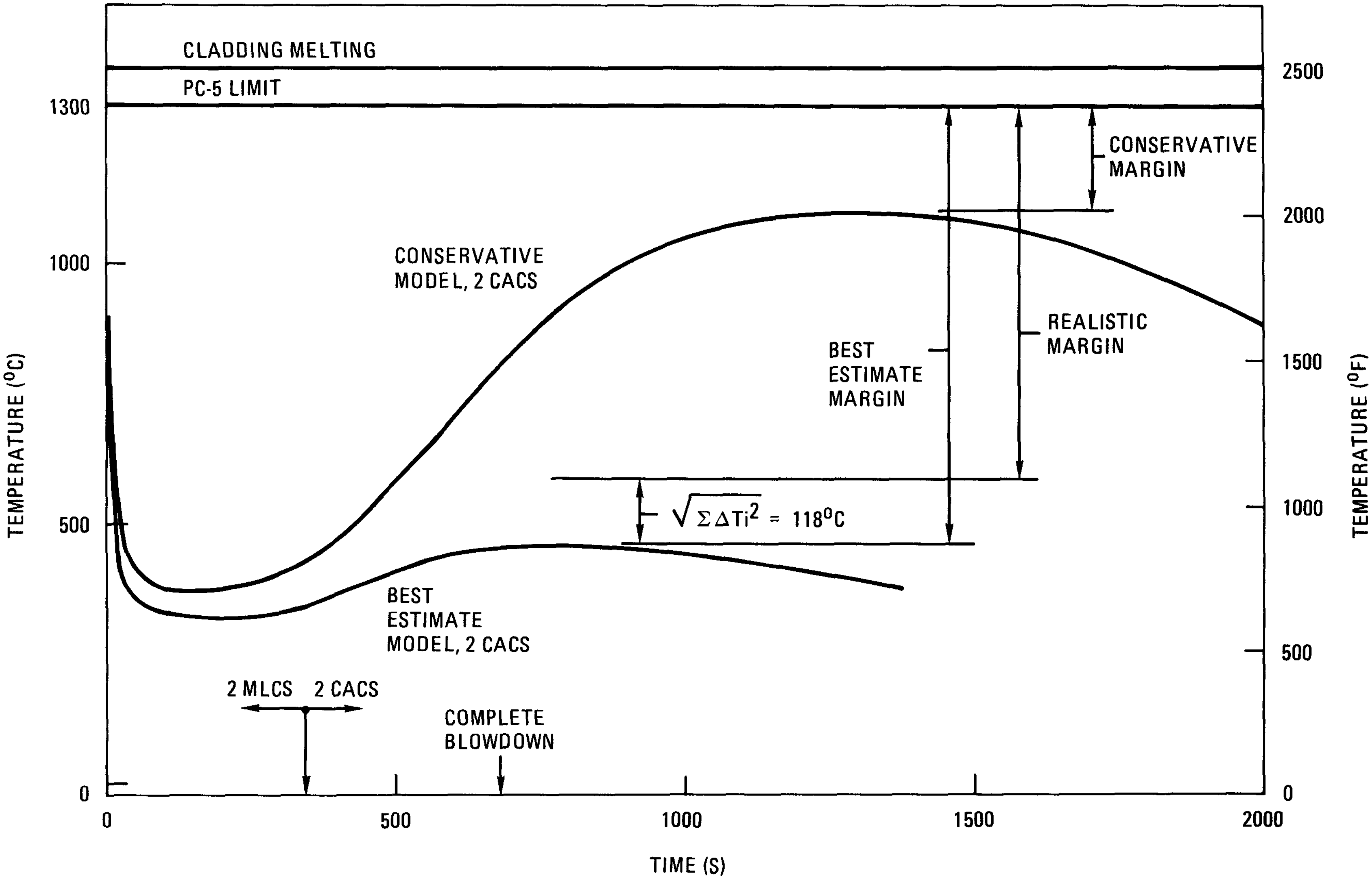

Fig. 5-39. Comparison of DBDA RHR performance margins

733°C (1319°F), indicating that the cumulative combination of all the uncertainty margins employed in the conservative model results in a large uncertainty penalty.

Obviously, a realistic approach is to combine the uncertainties statistically. To evaluate sensitivity, each of the uncertainties is applied individually to the best-estimate model. Table 5-15 shows the results of the sensitivity analysis. Uncertainties of significant effect are +20% decay heat, +20% loop flow resistance, -17% back pressure, and -15% overall conductance in heat exchangers (the CAHE and the ALC). The arithmetic sum of the deviations is 271°C (488°F), which is much smaller than the deviation due to cumulative uncertainties [i.e., 733°C (1319°F) peak cladding temperature difference between the conservative and the best-estimate models]. This indicates the effect of nonlinearity in the correlations.

By examining closely for the cause of this large discrepancy, it is known that the core flow is fully turbulent in the best-estimate case and in all the perturbation cases with individual uncertainty, while the core flow in the conservative model is laminar because of cumulative combination of uncertainties in the worst direction. The heat transfer and the friction correlations for the two flow are significantly different and result in such a large discrepancy.

The root-sum-square of the deviations represents the response uncertainty of same band width as the individual uncertainty bands , if the variables are assumed to be mutually independent and linear, the error distributions are assumed to be symmetrical, and the probability of the overall uncertainty is assumed to be equal to the individual uncertainties.

This area of the uncertainty treatment is not well defined and is undergoing further industry-wide development for alternative approaches. The root-sum-square is assumed herein as a guide to the realistic safety margin.

TABLE 5-15
SENSITIVITY OF THE DBDA CORE CLADDING TEMPERATURE
TO INDIVIDUAL PARAMETER UNCERTAINTIES

Parameter	Uncertainty Margin (%)	Maximum Fuel Cladding Temperature Deviation, ΔT_i [°C(°F)]
Initial Power	+2	6 (1319)
Decay heat	+20	92 (165)
Local power	+5	19 (34)
Primary, secondary, and tertiary loop flow resistance	+20	23 (41)
Overall conductances in CAHE and in ALC	-15	35 (63)
Primary coolant thermal conductivity	-5	14 (25)
Primary coolant viscosity	+3	7 (13)
Containment backpressure	-17	49 (88)
Channel enthalpy rise	+11	27 (49)
Arithmetic sum $\sum_{i=1}^{9} \Delta T_i$		271 (488)
Statistical sum $\sum \Delta T_i 2$		118 (212)
Cumulative uncertainty (all uncertainties applied simultaneously)		733 (1319)

Figure 5-39 shows a perspective of the type of safety margins which are obtained by different uncertainty treatments. The conservative model with cumulative uncertainties shows the conservative safety margin of 200°C (360°F) to the design limit. This margin is adequate to allow for the local temperature anormalties, such as the fuel assembly edges and corner effects (see Section 5.1.2.1 and Appendix B). The statistical uncertainty combination results in the realistic safety margin of 719°C (1294°F) to the design limit. The best-estimate model without system parameter uncertainties gives the best-estimate margin of 837°C (1507°F). An even higher margin results in the case of the best-estimate model with three CACS loops. This would ordinarily be available without a single failure.

Therefore, a sufficient margin of safety is concluded to be provided in the RHR system performance under the most limiting set of core cooling conditions (i.e., DBDA).

5.7.2. RHR Redundancy Margin

In the previous sections on RHR performance evaluation, limiting sequences of events are selected and analyzed according to the deterministic safety evaluation rules of ANS-50, Policy 2.4 (Ref. 5-2) (see Section 5.1.1 and Appendix A). In these event sequences, adequate RHR is found to be achieved without using many of the functionally redundant RHR systems. The performance adequacy of these redundant systems is demonstrated using margin cases which are defined by assuming multiple failures beyond the deterministic safety requirement.

To provide a perspective for the depth of protection with redundant RHR systems, Figs. 5-40 and 5-41 and Table 5-5 summarize the results of all the cases analyzed. Figures 5-40 and 5-41 show the maximum fuel and blanket cladding temperatures, respectively, with the abscissa indicating the RHR system loop type, such as MLCS, SCS, CACS, or natural circulation. The dark symbols signify the margin cases. Table 5-5 lists notation for the symbols,

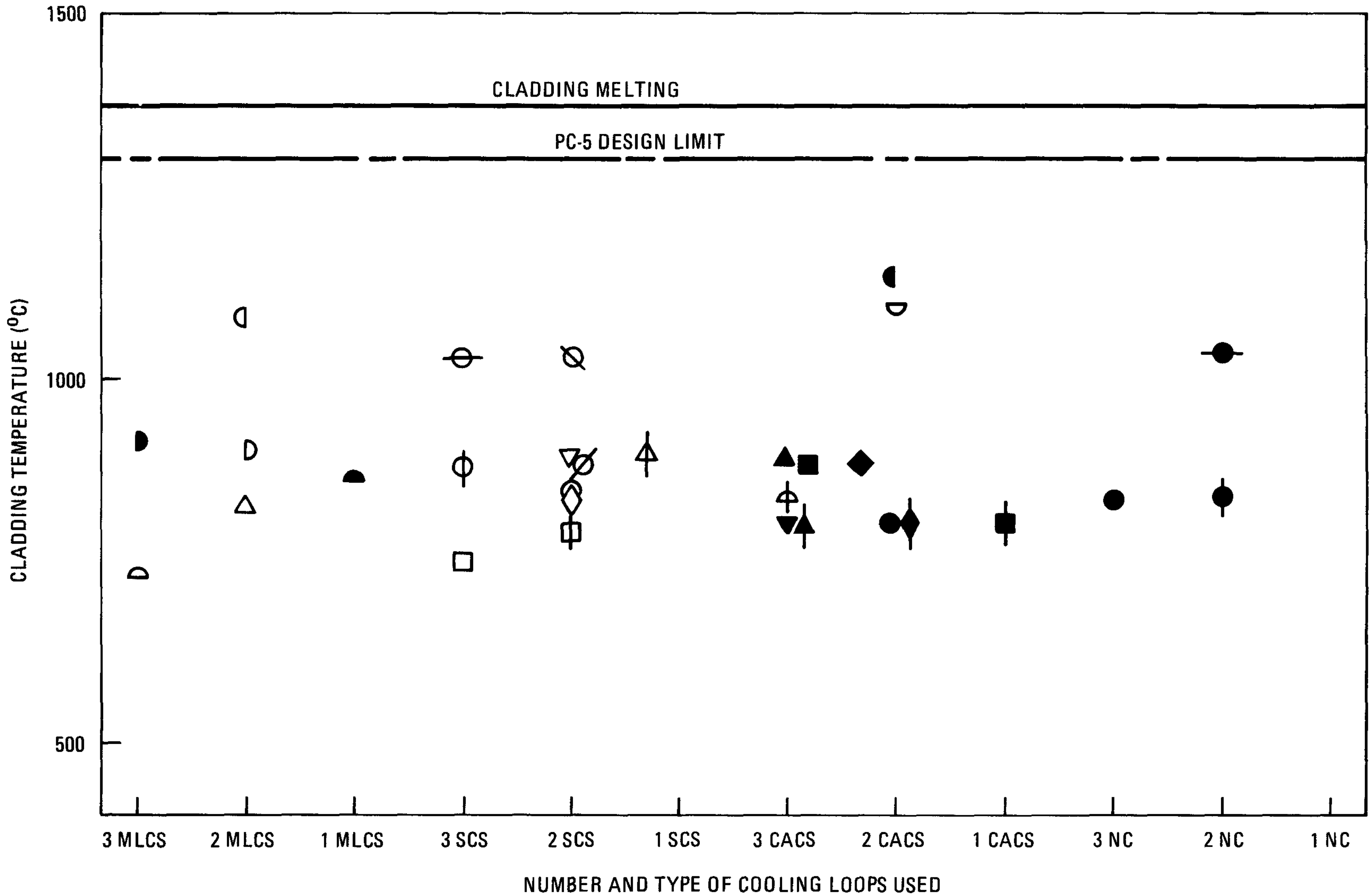

Fig. 5-40. Summary of core cooling performance; maximum fuel cladding temperatures
Note: Table 5-5 gives captions for symbols.

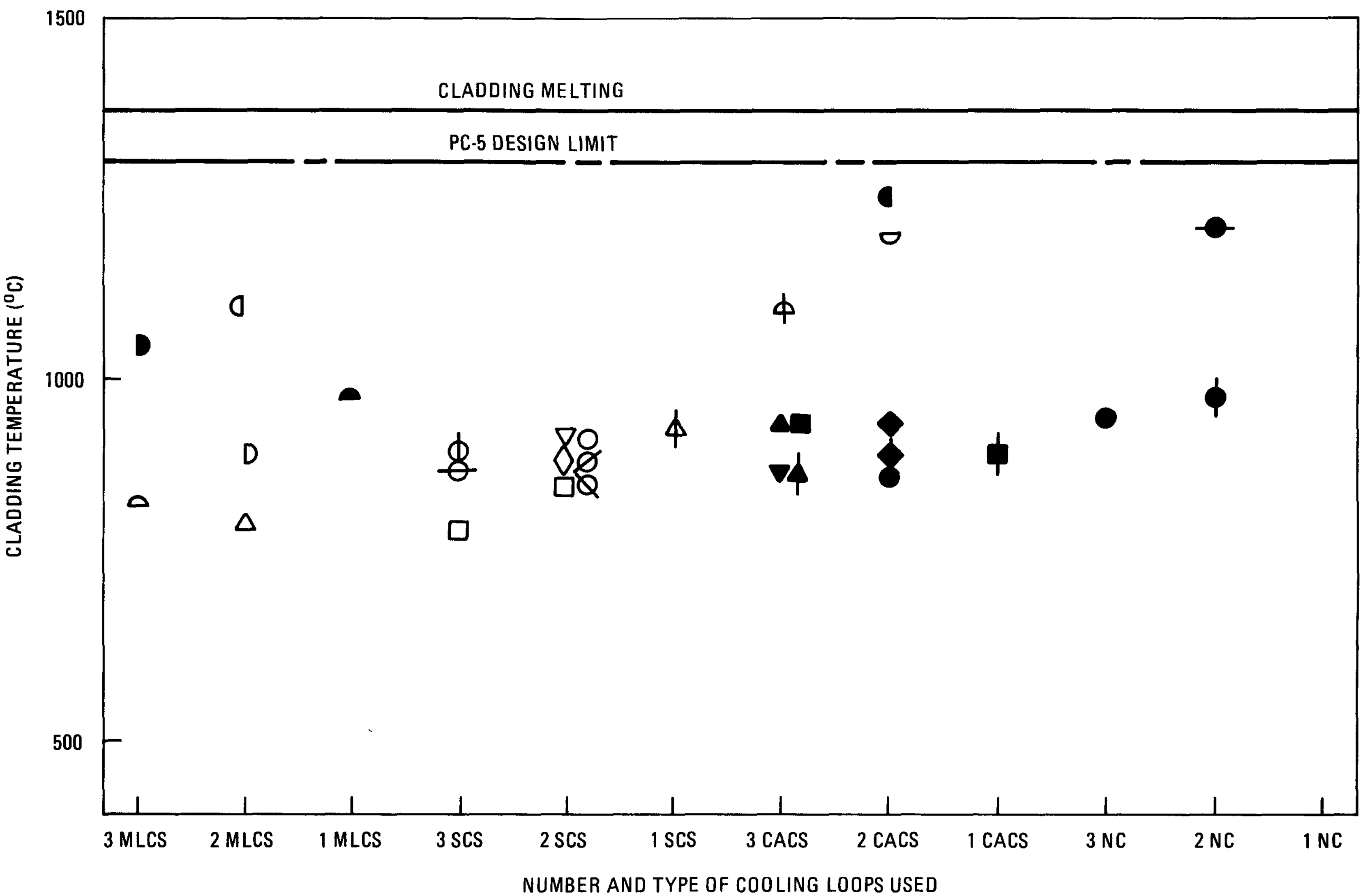

Fig. 5-41. Summary of core cooling performance; maximum blanket cladding temperatures
Note: Table 5-5 gives captions for symbols.

including type of events, assumed failures, relevant PCs, the RHR system used, maximum fuel and blanket cladding temperatures with their applicable limits, and the peak reactor outlet plenum temperature with the thermal barrier temperature limits. Table 5-5 indicates how well the maximum temperatures affecting the critical components meet their limits.

From this analysis summary, a large margin of safety can be concluded to be provided in the GCFR RHR capability as a whole.

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6. ELEMENTS OF A PLAN FOR GCFR RESIDUAL HEAT REMOVAL AND NATURAL CIRCULATION VERIFICATION

The gas-cooled fast-breeder reactor (GCFR) features three residual heat removal (RHR) systems: (1) the main loop cooling system (MLCS), (2) the shutdown cooling system (SCS), and (3) the core auxiliary cooling system (CACS). These systems are initiated by the plant protection system (PPS). In the primary mode of operation for these systems, heat is removed from the core and transferred to the ultimate sink via primary, secondary, and tertiary loops using forced circulation. Additionally, the CACS design incorporates natural circulation cooling capability as a backup to the forced circulation modes.

This section outlines key elements of two separate plans to verify each operation mode. Section 6.1 addresses verifying GCFR-RHR capability; it focuses on the deterministic requirements that are met by forced circulation RHR systems. Section 6.2 describes a similar plan developed to verify and validate the GCFR natural circulation RHR capability, which is considered to provide an added margin for events beyond the design basis.

6.1. RHR VERIFICATION PLAN

This section describes the GCFR RHR performance verification plan. The plan objective is to define a set of tasks that will develop confidence in the RHR system design and its performance by verifying and validating the predictive methods used to develop that design.

The predictive methods used in the design of the GCFR-RHR system are computer codes (such as FASTRAN) that represent the design via a collection of component models. These models are typically sets of differential and algebraic equations that the codes solve to predict the system temporal behavior. The adequacy of RHR system behavior is judged by evaluating an RHR performance measure (such as the difference of the predicted fuel cladding temperature from its design limit value). The plan elements have been designed to focus verification and validation efforts only on those models that most impact the RHR performance measures.

RHR performance verification plan elements are listed below in their proposed order of execution:

1. Survey RHR systems and performance verification for other fast reactors.

2. Identify key issues for GCFR RHR system performance.

3. Verify by comparing to independent codes.

4. Validate using data from other reactors.

5. Validate by component and subsystem tests.

6. Validate by preoperational and startup tests.

7. Investigate RHR system adequacy for postulated event sequences.

The following subsections outline the philosophy and function of each plan element.

6.1.1. Survey RHR Systems and Performance Verification for Other Reactors

The function of this plan element is to reveal the comprehensiveness of RHR system designs and verification plans proposed for other reactors. A detailed literature survey should determine and describe the following:

1. Plans to verify RHR component performance.

2. Comprehensive failure modes assumed in developing the RHR systems and their control/protection strategies.

The objective of this literature review is to guide GCFR RHR verification plan by using plans developed for other reactors, such as the liquid metal fast breeder reactor (LMFBR).

6.1.2. Identify Key Issues Pertinent to RHR System Performance

RHR system performance adequacy is determined by how well the RHR system removes decay heat from the core to the ultimate heat sink. The RHR system performance is adequate if it meets the GCFR component design limits.

Table 6-1 shows issues and variables that influence RHR system performance. The issues are based on RHR performance sensitivity experience gained by using FASTRAN (Ref. 6-1). Clearly, the thermal-hydraulic performance of the individual RHR system components is important. The PPS is crucial, because it activates the RHR systems. The plant control system (PCS) is important, because it modulates the behavior of coolant system components before they are placed in the RHR mode. The adequacy of these responses is strongly influenced by the disturbance (event sequence) hypotheses that are adopted for PPS design.

6.1.3. Verification by Comparison to Independent Codes

The objective of this plan element is to verify selected computer codes by comparing them with similar independently developed codes used in the nuclear industry. This element investigates only those key issues for which a code-to-code comparison will resolve modeling or solution uncertainties. General Atomic proposes to emphasize verifying FASTRAN code segments dealing with the core, steam generator, and core auxiliary heat exchanger (CAHE). A special verification effort is planned for the entire CACS system, because its nature is crucial as a final line of defense in the RHR hierarchy. General Atomic is working on phases of this effort via a subcontract to EG&G of Idaho. General Atomic intends to have EG&G independently review the entire GCFR-RHR design analysis. This review will produce a new GCFR plant

TABLE 6-1
KEY ISSUES INFLUENCING RHR PERFORMANCE MEASURES

<u>RHR component performance</u>

- Circulator performance
  - Circulator performance characteristics
  - Transient response
  - Parallel operation
  - Stall margin during depressurization
- Valve performance
  - Pressure difference versus flow
  - Friction
- Core performance
  - Thermal
  - Hydraulic
  - Interassembly and intra-assembly flow distribution
  - Edge channel performance
- Loop flow resistance
  - Primary
  - Secondary
- Heat exchanger performance
  - Steam generator
  - Core auxiliary heat exchanger (CAHE)

<u>Control and protection system performance</u>

- PPS induced hierarchy
  - Sequencing produced by PPS, bridging or skipping problems
  - Impact of malfunctions
  - Interaction of subsystems during normal PPS controlled transition
- Implementation problems
  - Setpoints, PCS
  - Gain settings, PCS
  - Thresholds, PPS

TABLE 6-1 (Continued)

Sensor errors

Actuator errors

Disturbance hypotheses

Design basis event sequences

RHR system response to high-probability, low-consequence events

Human factors/operational procedures

Manual overrides of the PPS

Maintenance or other procedures, assess compatibility with RHR operational assumptions simulation that can verify the FASTRAN code at the system level. This verification effort will independently develop and compare the following models:

1. GCFR core.

2. MLCS components (steam generator, feed pump, main circulator/ motor.

3. SCS components (pumps, condenser, main circulator/pony motor).

4. CACS components [auxiliary circulator, CAHE pump, auxiliary loop cooler (ALC)].

5. PPS and PCS logic and loops.

The FASTRAN core model will be verified both by comparing it to a similar EG&G model and by comparing it to the more detailed COBRA-IV subchannel thermal-hydraulic analysis code. COBRA-IV will be verified by comparing it to other subchannel analysis codes available in the nuclear industry.

6.1.4. Validation Using Data from Other Reactors

This plan element validates those issues identified in Section 6.1.2 using data from other reactors. Issues selected for this type of validation are (1) primary coolant valve performance and (2) steam generator performance. Data are available for both of these from Fort St. Vrain tests of very similar components.

6.1.5. Validation by Component and Subsystem Tests

This plan element validates code component and subsystem models using data from tests of actual system components. These data will be collected in component tests and subsystem test facilities. The major subsystem test facilities planned are (1) the circulator test facility, (2) the core flow test loop (CFTL) (to be located at Oak Ridge National Laboratory), and (3) the primary coolant flow test facility (to be located at General Atomic). The function of each is discussed below.

6.1.5.1. Test Facilities.

Circulator Test Facility. The circulator test facility will test a number of demonstration plant prototype and production circulator components (i.e., the main and auxiliary circulators, their drives, controllers, service system, and isolation valves). These tests are to be performed over the entire range of pressure, temperature, helium flow rate, and rotating machinery speed as expected in the demonstration plant.

Core Flow Test Loop. A series of out-of-pile tests will confirm the predicted performance of each GCFR core assembly type by simulating the dynamic reactor environment in a large helium loop with assemblies of electrically heated rods. Specifically, the tests will verify the design, explore for design deficiencies, check the results of other GCFR development program tests over a range of design and postulated accident conditions. Safety margins under extreme undercooling, overcooling, overpower, and depressurized conditions will be determined.

Primary Coolant Flow Test Facility. The primary coolant flow test facility will be a one-third scale model of the primary flow paths (1) from the core outlet plenum to the steam generator inlet and (2) from the cold ducts through to the inlet plenum. This facility will primarily evaluate the effects of complicated geometry not amenable to analysis. Factors being examined for path 1 above will be (1) hot streaks, (2) flow distribution, and (3) differential pressure versus flow correlations. For path 2 above, the main interest will be in hot streaks.

6.1.5.2. Validation Efforts. Data collected from these test facilities will be used to resolve uncertainties in circulator performance, valve performance, core performance, and loop flow resistance, as described below.

Circulator Performance.

Circulator Maps. The first set of actual circulator test data will be obtained at the circulator test facility when it is used as a test bed for the prototype circulator. During these tests, the performance maps for forward and reverse flow will be generated for all speeds in the RHR operating range. These maps will validate the performance maps used in FASTRAN circulator models. Particular attention will be focused on surge margins.

Transient Response. The circulator test facility also provides an opportunity to gather transient data on an actual circulator operating in a representative environment. Startup and coastdown transients from this facility will estimate characteristic values of the circulator as its operating point varies with RHR condition. This information will validate FASTRAN models of circulator load torques.

Primary Coolant Valve Performance.

Pressure Difference Versus Flow. The variation of pressure drop with flow through the isolation valves will be validated by testing an actual isolation valve at the circulator testing facility.

Friction. The friction levels associated with opening and closing the isolation valves will be validated using data from the circulator test facility.

Core Performance.

Thermal-hydraulics. The CFTL will allow detailed (subchannel level) thermal-hydraulic data to be collected on GCFR fuel, blanket, and control

Core Performance.

Thermal-hydraulics. The CFTL will allow detailed (subchannel level) thermal-hydraulic data to be collected on GCFR fuel, blanket, and control assemblies. These data will validate the predictions by codes, such as COBRA-IV. These data will also indirectly validate the FASTRAN core model via the verification efforts described in Section 6.1.3.

Loop Flow Resistance.

Primary Loop. General Atomic plans to develop a facility to study the characteristics of the primary coolant flow. The facility will emphasize determining flow streaking and maldistribution phenomena that may occur in the primary loop. Air will be a test fluid. Data will also be collected on the distribution of pressure drop around this test loop. These data will serve as a basis to validate FASTRAN primary loop assumptions.

6.1.6. Validation by Preoperational and Startup Tests

A number of validation tests can be accomplished in facilities incorporated in GCFR development plans. Most circulator and isolation valve issues can be resolved using data from the circulator test facility. Questions regarding subchannel effects in core fuel, blanket, or control assemblies can be resolved using CFTL data. Several validation issues can be resolved at the component test level. Many of the remaining issues for the validation efforts will be cleared during the demonstration plant preoperational or startup tests.

Many validation efforts are left to this point, because the design uncertainty design risk is too small to justify the cost of a special test facility. Some efforts are postponed, because the necessary combination of components or subsystems are not suitably integrated in the appropriate environment until the demonstration plant phase.

The startup and preoperational validation tests identified in this plan element fit one of these categories. This plan element will determine functional adequacies in the following areas:

1. Circulator performance.
2. Circulator transient behavior.
3. Isolation valve friction characteristics.
4. Plenum to plenum core pressure drop.
5. Primary system pressure drop.
6. Core intra-assembly flow distribution.
7. Overall primary and secondary loop flow characteristics.
8. Heat exchanger performance.
9. RHR system selection logic.

The tests will be performed in a sequence that will preclude damages to the core and the plant hardware. The test conditions may include zero power, non-nuclear heating from the circulator work, and low power in the range of 0 to 20% under pressurized and depressurized conditions.

6.1.7. Investigate RHR System Adequacy for Postulated Event Sequences

The verification and validation efforts discussed in previous plan elements concentrate primarily on RHR system component performance. As pointed out in Section 6.1.2, the performance of the control/protection systems and the influence of the assumed accident sequences are also key elements in RHR performance. FASTRAN predictions need to be verified for the proposed PPS strategy, RHR loops, and possible accidents. General Atomic proposes to develop an analog/hybrid simulation that will operate in a scaled-time mode for testing and analysis verification. This simulation will contain the following:

1. A core model.

2. MLCS components (steam generator, main circulator, feed pump) model.

3. SCS components (pumps, condenser, main circulator/pony motor) model.

4. CACS components (auxiliary circulator, CAHE, pump, ALC) model.

5. PPS (logic, sensors, actuators) model.

6. PCS (loops, sensors, actuators) model.

The initial simulation purpose will be to check the RHR system response to the design basis event sequences that are used in developing PPS-RHR strategies. After thoroughly checking the system performance against these event sequences, variations of these sequences will be generated to test the RHR system adequacy.

A major concern since the Three Mile Island nuclear incident is the need to evaluate protective system responses to high probability/low consequence event sequences and human errors in complying with operational procedures. General Atomic feels that the interactive capability provided through a properly developed analog/hybrid simulation could effectively develop compatible maintenance/ operator procedures and RHR systems.

6.2. NATURAL CIRCULATION VERIFICATION PLAN

This section outlines the key plan elements for verifying the natural circulation capability of the upflow GCFR design. The main objective of a natural circulation verification plan is to demonstrate by analysis and testing that the candidate GCFR design can provide adequate natural circulation cooling. The plan elements outlined in this section describe a sequence of efforts that will, when fully defined and executed, verify natural circulation capability in a stepwise fashion.

The natural circulation verification plan elements are described by the following section titles:

1. Literature survey of natural circulation in other reactors.
2. Key mechanisms influencing natural circulation in the GCFR.
3. Verification by comparison to independent codes.
4. Validation by comparison to experimental data.
5. Validation by component tests.
6. Validation by scale model tests.
7. Validation by preoperational and startup tests.

The remainder of Section 6 describes the function of each plan element and sufficiently details each key plan element to show the scope of the required effort.

6.2.1. Literature Survey of Natural Circulation in Other Reactors

Information obtained from a detailed literature survey of natural circulation phenomena, analyses, testing, and verification efforts for light water reactors (LWRs), LMFBRs, and gas-cooled reactors (GCRs) will provide the following:

1. Guidance in identifying key phenomena influencing natural circulation in the GCFR and in developing a natural circulation verification plan for the GCFR.

2. A data base of industry codes and experimental data to be used in verification of GCFR codes.

6.2.2. Key Mechanisms Influencing Natural Circulation

Table 6-2 gives a preliminary list of important mechanisms influencing natural circulation in the GCFR. Detailed sensitivity analyses of these variables will be performed to aid in defining natural circulation test

TABLE 6-2
A PRELIMINARY LIST OF KEY MECHANISMS INFLUENCING NATURAL CIRCULATION IN THE GCFR

Transition from forced circulation to natural circulation in the primary system, including the startup of natural circulation from essentially stagnant conditions.

Circulator coastdown.

Primary system flow resistance (core, heat exchangers, locked auxiliary circulator rotor, valves, loop).

Secondary and tertiary system flow resistance.

Primary, secondary, and tertiary system heat capacities, heat losses, and effect of heat transfer on temperature profiles.

Mixing, potential flow stratification, cold traps, and local recirculation effects.

Thermal center effect in steam generator.

Fuel and blanket heat transfer coefficients and friction factors under natural circulation conditions.

Interassembly and intra-assembly core flow redistribution.

Main and auxiliary loop valve actuations, malfunctions, and loop isolation.

Primary, secondary, and tertiary system boundary conditions at onset of natural circulation.

requirements and verification efforts. In arriving at the list of key mechanisms, the following two points are important:

1. Natural circulation is driven by the integrated temperature profile in the primary, secondary, and tertiary loops and depends on all the thermal-hydrualic characteristics of these systems, including local effects that may not be important during forced circulation.

2. The same basic physical phenomena apply to both steady-state performance and to the transition from forced to natural circulation. However, the uncertainty in modeling complex and potentially localized thermal-hydraulic effects using one-dimensional system codes is greatest during the transient phase of natural circulation. These uncertainties include establishing the proper temperature profile at the onset of natural circulation in all loops, transfering from the main to auxiliary loops via passive valve actuation, and starting primary coolant natural circulation in the auxiliary loops from essentially stagnant conditions.

6.2.3. Verification by Comparison to Independent Codes

The objective of this plan element is to verify all computer codes used in safety analysis of GCFR natural circulation by comparing them with similar codes used in the nuclear industry.

The GCFR computer codes to be verified in this manner include the following:

1. System codes RATSAM and FASTRAN (see Sections 5.1.6.1 and 5.1.6.2).

2. The FASTRAN and RATSAM core thermal model and the FASTRAN CACS model.

3. COBRA, the core subassembly thermal-hydraulic analysis code.

6.2.4. Validation by Comparison to Experimental Data

Another means of validating GCFR natural circulation codes is by comparing it with existing experimental data obtained from natural circulation tests involving GCRs and LWRs.

6.2.4.1. Primary System, Sizewell. A comparison of data obtained from the Central Electric Generating Board (CEGB) for natural circulation tests, conducted at their Sizewell Nuclear Power Station in Great Britain, and RATSAM predictions for the test will validate the capability of the one-dimensional system code to calculate the transition of natural circulation in a GCR following reactor trip and circulator coastdown. Sizewell is a CO_2-cooled MAGNOX reactor.

6.2.4.2. Secondary System, LWR. The CACS secondary loop is similar to a pressurized water reactor (PWR), since boiling is suppressed. General Atomic is investigating the possibility of using PWR data to validate the CACS secondary loop model in FASTRAN. These data are potentially available from two sources:

1. PWR natural circulation tests performed during plant startup, as required by Regulatory Guide 1.68 (Ref. 6-2).

2. PWR natural circulation tests scheduled to be performed in the next few months at experimental facilities FLECHT and LOFT. Because of extensive instrumentation of the test loops, these data are expected to be more comprehensive and complete for use in validation efforts than data available from plant startup tests.

6.2.5. Validation by Component Tests

The component test results will validate computer models and design performance of critical components under natural circulation conditions. Component tests of the core, main, and auxiliary circulators and the main and auxiliary loop isolation valves (MLIV and ALIV) for RHR verification (Section 6.1.5) are sufficient to validate natural circulation, but must be performed under conditions representative of natural circulation. Critical component tests required for validating natural circulation include the following:

1. Measure ALIV leakage flow as a function of the valve differential pressure prior to transfer to CACS natural circulation.

2. Measure flow resistance across the nonrotating auxiliary circulator impeller.

3. Measure core intra- and interassembly flow redistribution effects, pressure drop components, and average heat transfer coefficient as a function of Grashof number and Reynolds number.

6.2.6. Validation by Scale Model Tests

As described in Section 6.2.2, natural circulation in the GCFR is a system-wide phenomena. Indeed, most of the uncertainties in modeling GCFR natural circulation performance using one-dimensional system codes relate to coupled system characteristics, including the transition from forced to natural circulation. Because of the need to resolve system uncertainties described in Table 6-2, scale model tests are currently being recommended by General Atomic to validate system codes (RATSAM and FASTRAN) and the GCFR natural circulation RHR system design for a specific scale model configuration. The results of sensitivity analyses will define the extent of system testing required.

6.2.6.1. Primary System Scale Model Test. The primary system test should include as much similitude to the GCFR as practical (i.e., approximately the same Reynolds number, Grashof number, time constants, transport times, system heat capacities, power gradients in core and radial blanket). To model the effect of outlet plenum mixing on the development of natural circulation, a heat source, consisting of several electrically heated fuel and radial blanket assemblies of differing power-to-flow ratios, can be used. A heat exchanger is required to simulate the CAHE. All duct geometries with bends in the circuit may have to be simulated to evaluate the potential for cold traps, flow stratification, and local recirculation.

The objectives of the primary system test are to determine the following:

1. System performance during the transition from forced to natural circulation, including the startup of natural circulation from stagnant (zero flow) conditions and steady-state system performance.

2. Effect of auxiliary loop boundary conditions (temperature profile and leakage flow prior to transfer to auxiliary loops) on the development of natural circulation.

3. Effect of fluid intertia, system heat capacities, mixing, potential flow stratification, cold traps, and local recirculation on the startup of natural circulation.

4. Intra-assembly flow redistribution effects upon natural circulation conditions.

5. Main and auxiliary loop interaction, including the failure of the main valve to close and/or the auxiliary valves to open.

6. System and core pressure drop and fuel and blanket cladding temperature response during various natural circulation conditions.

6.2.6.2. Secondary System Test. The objectives of the secondary system test are to determine the following:

1. Secondary system performance at steady state and during the development of natural circulation as a function of a wide variety of potential initial thermal and flow boundary conditions.

2. The effect of system heat capacities, fluid inertia, mixing, cold traps, and possible flow stratification on the development of natural circulation.

6.2.6.3. Tertiary System. The need to perform natural circulation tests for the tertiary system (i.e., water-to-air heat exchanger) will be evaluated. No tests are currently planned.

6.2.7. Validation by Preoperational and Startup Tests

Preoperational and startup tests will provide the ultimate validation of the GCFR demonstration plant natural circulation RHR system design and system code predictions.

6.2.7.1. No Power/Nonnuclear Heating Tests. General Atomic will investigate the possibility of no power/nonnuclear heating tests at pressure with the core installed but shut down and with the system heated [to 316°C (~600°F)] by main circulator compressive heating. This potentially allows two types of tests:

1. Trip main circulators, run auxiliary circulators, and remove heat from the primary system by natural circulation in the secondary and tertiary CACS loops.

2. Trip main circulators, transfer to an auxiliary loop(s) which has been maintained at a very low temperature by forced circulation cooling on the secondary side; upon transfer, monitor

the transition from essentially stagnant conditions to natural circulation in the cold auxiliary loop.

6.2.7.2. Low Power Test. The ultimate natural circulation design validation will be a low power test in the demonstration plant at 0% to 20% power. A sequence of tests (including tests before fuel loading) will resolve uncertainties in a progressive fashion without risk of damage to plant hardware. Extensive analysis prior to testing will insure that the combination of core inlet temperature, core power, and natural circulation flow will keep hot spot fuel and blanket clad temperatures below design operating levels (well below any damage limits). The final test(s) will be of the natural circulation design condition (albeit lower than design power level): reactor scram, main circulator coastdown, and transfer to CACS loops by gravity-actuated valve operation, followed by transition to natural circulation using the CACS loops.

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APPENDIX A
APPLICATION OF ANS-50, POLICY 2.4, DETERMINISTIC SAFETY RULES
TO GCFR RHR SYSTEMS

Tables A-1 through A-6 present the matrix method developed to evaluate the deterministic safety rules for gas-cooled fast breeder reactor (GCFR) core cooling systems (see Section 5.1.1). The rules are applied on a loop-by-loop/system-by-system basis as activated by the plant protection system (PPS) (see Notes for Appendix A).

Table A-1 presents an example of one initiating event, a trip of one main circulator. In the first case (CT-1), the circulator remains operational and thus can perform its shutdown cooling system (SCS) function. In cases CT-2 and CT-3, the circulator is assumed inoperative; thus, one SCS loop is also inoperative.

For Case CT-1:

Event CT-12M (see Note 5). One main loop cooling system (MLCS) loop is isolated due to the initiating event. Cooldown is on the remaining two MLCS loops, and plant condition-2 (PC-2) limits apply.

Event CT-13S (CT-12M + LOSP). A loss of offsite power (LOSP) is applied, isolating the remaining MLCS loops. The PPS activates the SCS (three loops) for cooldown, and PC-3 limits apply.

Event CT-12S (CT-13S + single failure). When the SCS starts, one loop fails to function (single failure). Cooldown is on the remaining two SCS loops, and PC-4 limits apply. With this event, the maximum limit change is reached (PC_{IE} + 2), and all the required postulated failures, consistent with initiating event assumptions, have been applied.

TABLE A-1
COMBINED EVENT FOR RHR(EXAMPLE)
(INITIATION EVENT & FAILURE SCENARIO)

INITIATING EVENT	PC	COMBINED EVENTS (FAILURE OR ACTION/PLANT CONDITION) [a] (COOLING SYSTEMS FAILED/OPERATING): 3 MLCS	1 MLCS \ 2 MLCS	2 MLCS \ 1 MLCS	3 MLCS \ 3 SCS	1 SCS \ 2 SCS	2 SCS \ 1 SCS	3 SCS \ 3 CACS	1 CACS \ 2 CACS	2 CACS \ 1 CACS	3 CACS \ 3 NC	1 NC \ 2 NC	2 NC \ 1 NC	OTHER FAILURES (SFO / COO)
		MAIN CIRCULATOR TRIP (CT)												
CT-1	2		IE \ 2		LOSP \ 3	SF \ 4								
CT-2 [b]	2		IE \ 2		LOSP \ (3)	IE \ 3	SF \ 4	MARGIN						
CT-3 [b]	2		IE \ 2		LOSP \ (3)	IE \ 4 SFO								MLIV OPEN

[a] SEE NOTES AT END OF APPENDIX A.

[b] ASSUMES CIRCULATOR INOPERATIVE.

TABLE A-2
CATEGORY 1
DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

INITIATING EVENT	PC	COMBINED EVENTS (FAILURE OR ACTION/PLANT CONDITION)[a] (COOLING SYSTEMS FAILED/OPERATING) 3 MLCS	2 MLCS / 1 MLCS	1 MLCS / 2 MLCS	3 SCS / 3 MLCS	2 SCS / 1 SCS	1 SCS / 2 SCS	3 CACS / 3 SCS	2 CACS / 1 CACS	1 CACS / 2 CSCS	3 NC / 3 CACS	2 NC / 1 NC	2 NC	OTHER FAILURES (SFO / COO)
CIRCULATOR TRIP	2		2 IE		3 LOSP	4 SF	MARGIN							
LOOP[b] TRIP	2		2 COO IE											REACTOR POWER REDUC-TION
CIRCULATOR BEARING SEIZURE	3		3 IE		(4) LOSP	4 IE	5 SF							
CIRCULATOR BEARING SEIZURE	3		3 IE		(4) LOSP	5 SFO IE								MLIV OPEN

(a) SEE NOTES AT END OF APPENDIX A.

(b) ATTEMPTS CONTINUED PLANT OPERATION ON REMAINING LOOPS AT REDUCED POWER.

TABLE A-3
CATEGORY 2
DECREASE IN CORE HEAT REMOVAL BY SECONDARY SYSTEM

		COMBINED EVENTS (FAILURE OR ACTION/PLANT CONDITION) [a] (COOLING SYSTEMS FAILED/OPERATING)												OTHER FAILURES
INITIATING EVENT	PC	3 MLCS	2 MLCS / 1 MLCS	1 MLCS / 2 MLCS	3 SCS / 3 MLCS	2 SCS / 1 SCS	1 SCS / 2 SCS	3 CACS / 3 SCS	2 CACS / 1 CACS	1 CACS / 2 CACS	3 NC / 3 CACS	2 NC / 1 NC	1 NC / 2 NC	(SFO COO)
LOSS OF CONDENSER VACUUM	2				2 / IE	3 / SF	← MARGIN	MARGIN	MARGIN	MARGIN	MARGIN	MARGIN	MARGIN →	
LOSS OF CONDENSER VACUUM (b), (c)	2				3 LOSP / IE			3 / LOSP	4 / SF					
LOSS OF CONDENSER VACUUM (b), (c)	2				3 LOSP / IE			4 SFO / LOSP						MLIV OPEN

(a) SEE NOTES AT END OF APPENDIX A.

(b) SCS RESTART FROM LOSP.

(c) DURING SCS RESTART, A PPS OR OPERATOR TRANSFER TO CACS OCCURS.

TABLE A-4
CATEGORY 3
DECREASE IN REACTOR COOLANT INVENTORY

		COMBINED EVENTS (FAILURE OR ACTION/PLANT CONDITION) [a] (COOLING SYSTEMS FAILED/OPERATING)													OTHER FAILURES
INITIATING[c] EVENT	PC	3 MLCS	1 MLCS / 2 MLCS	2 MLCS / 1 MLCS	3 MLCS / 3 SCS	1 SCS / 2 SCS	2 SCS / 1 SCS	3 SCS / 3 CACS	1 CACS / 2 CACS	2 CACS / 1 CACS	3 CACS / 3 NC	1 NC / 2 NC	2 NC / 1 NC		(SFO COO)
DBDA (30 IN.^{2})	5	5			LOSP / NDR			NDR / 5	SF / 5						
DBDA (30 IN.^{2})	5		IE / 5		LOSP / NDR			NDR / 5 SFO	← MARGIN →						MLIV OPEN
DBDA (30 IN.^{2})	5		IE / 5 SFO												MLIV OPEN
DBDA (30 IN.^{2}) (b)	5									← NOT DESIGNED TO RESPOND →					
DBDA (100 IN^2)	5														

[a] SEE NOTES AT END OF APPENDIX A.

[b] NATURAL CIRCULATION IN CACWS (AIR AND WATER).

[c] ALL EVENTS INCLUDE AN INGRESS OF AIR-HELIUM MIXTURE FROM CONTAINMENT.

TABLE A-5
CATEGORY 4
REACTIVITY ACCIDENTS

		COMBINED EVENTS (FAILURE OR ACTION/PLANT CONDITION) [a] (COOLING SYSTEMS FAILED/OPERATING)												OTHER FAILURES
INITIATING EVENT	PC	3 MLCS	1 MLCS / 2 MLCS	2 MLCS / 1 MLCS	3 MLCS / 3 SCS	1 SCS / 2 SCS	2 SCS / 1 SCS	3 SCS / 3 CACS	1 CACS / 2 CACS	2 CACS / 1 CACS	3 CACS / 3 NC	1 NC / 2 NC	2 NC / 1 NC	(SFO / COO)
CONTROL ROD WITH-DRAWAL (@ 100%)	3	3			LOSP / 4	SF / 5								
CONTROL ROD WITH-DRAWAL (@ 25%)	3	3			LOSP / 4	SF / 5	MARGIN							

(a) SEE NOTES AT END OF APPENDIX A.

TABLE A-6
CATEGORY 5
LOSS OF FORCED CIRCULATION

		COMBINED EVENTS (FAILURE OR ACTION/PLANT CONDITION) [a] (COOLING SYSTEMS FAILED/OPERATING)												OTHER FAILURES
INITIATING EVENT	PC	3 MLCS	2 MLCS / 1 MLCS	1 MLCS / 2 MLCS	3 SCS / 3 MLCS	2 SCS / 1 SCS	1 SCS / 2 SCS	3 CACS / 3 SCS	2 CACS / 1 CACS	1 CACS / 2 CACS	3 NC / 3 CACS	2 NC / 1 NC	1 NC / 2 NC	(SFO COO)
LOSS OF ALL AC POWER	5	←			NON-DETERMINISTIC					→	5 / IE	5 / SFO		ALIV CLOSED
LOFC DURING REFUELING[b]	5	←			NON-DETERMINISTIC					→	5 / IE	5 / SFO		ALIV CLOSED
											←	MARGIN		→

(a) SEE NOTES AT END OF APPENDIX A.

(b) DURING REFUELING. REPRESSURIZATION OF PCRV REQUIRED.

For the assumption of an inoperative circulator as the initiating event, two cases are necessary (CT-2 and CT-3) to evaluate all the required postulated failures.

The remaining cooling loops or systems which have not been failed or activated, respectively, represent a margin in core cooling capability. For this initiating event, the margin consists of forced circulation [on the core auxiliary cooling system (CACS)] followed by natural circulation (NC) on the CACS loops.

From this set of combined events generated from one initiating event, limiting event candidates are selected by comparative evaluation of available heat removal capability versus the design or safety limits for the various plant systems and components. Analyses are then performed on the selected events to determine the limiting events.

This procedure was applied to all initiating events, and Tables A-2 through A-6 present the selected limiting and nonlimiting candidate cases according to category (see Section 5.1.1). The dashed boxed areas are those events whose analyses are presented and discussed in Section 5, and they include margin events beyond the requirements of the deterministic safety rules.

Except for Category 3, the reactor remains pressurized. An abundant margin in core cooling capability exists from both forced and natural circulation. In the second and third loss-of-condenser-vacuum cases in Category 2, the required LOSP has been applied during SCS startup. The LOSP is assumed to cause the SCS startup sequencer to trip, necessitating a reset of the sequencer and, thus, an SCS restart.

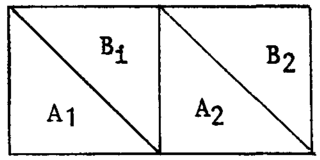
However, for a margin case during this delay, a PPS or an operator-initiated startup of the CACS is assumed to occur (cancelling the SCS startup sequencer). The remaining required postulated failures are then applied; this extends the case into the margin area.

In Category 3, decrease in reactor coolant inventory, natural circulation on the CACS is inadequate with the reactor depressurized. Moreover, for leak areas $\geq$200 cm^2 ($\geq$30 in.^{2}), cooldown by forced circulation on one CACS loop is not adequate. As a result, some failure scenarios leave no margin in core cooling capability, as indicated by the first case.

In Category 5, loss of forced circulation, the MLCS, SCS, and CACS are assumed to fail to function due to a loss of all ac power, and cooldown is by natural circulation on the CACS loops. This requires a failure scenario extending beyond the requirements of the deterministic safety rules; thus, any case in this category is considered a margin case. In the first case, the reactor is initially operating and remains pressurized. In the second case, the reactor is shut down for refueling and must be repressurized before natural circulation can occur.

NOTES FOR APPENDIX A

1. Legend:



A_1	failure or action rendering this loop(s) inoperative
B_1	resulting plant condition (design limits) with plant operating on next loop(s) in preferred sequence
A_2	next failure or action in failure scenario
B_2	See B_1.
(B_i)	intermediate step in failure scenario, a nonevent

2. The following symbols are used:

PC	plant condition
MLCS	main loop cooling system
SCS	shutdown cooling system
CACS	core auxiliary cooling system
NC	natural circulation on CACS loops
SF	single failure
SFO	single failure in other systems
COO	coincident occurrence in other systems
LOSP	loss of off-site power
IE	initiating event
NDR	not designed to respond
S-R	safety-related
P	probability of coincident occurrence
MLIV	main loop helium isolation valve
BFP	boiler feedpumps
ALIV	CACS loop helium isolation valve
PPS	plant protection system
TT	trip of turbine-generator

3. LOSP as a noninitiating event assumes loss of main circulator power (tripping loops) plus trip of BFP.

4. All events assume a reactor trip plus turbine trip and main circulator coastdown unless noted otherwise.

5. Event identification is by table (matrix) coordinates:

Event	Coordinates
CT12M	CT-1 and two MLCS loops operating
CT13S	CT-2 and three SCS loops operating

6. All events assume the maximum worth control rod stuck in full withdrawn positon.

APPENDIX B
TWO-DIMENSIONAL FUEL AND BLANKET ASSEMBLY ANALYSIS

B.1. INTRODUCTION

The residual heat removal (RHR) analyses presented in the main text were conducted with the FASTRAN computer code, which calculates the expected transient response of the gas-cooled fast breeder reactor (GCFR) core blanket assemblies during selected event sequences. The FASTRAN code employs a unit cell model of the multirod core assemblies and, thus, does not include the effects of intersubchannel flow mixing, circumferential heat conduction, or heat radiation.

When these two-dimensional effects are accounted for, under accident conditions, maximum cladding temperature in the grid-spaced fuel assembly tends to occur locally at the assembly edge adjacent to the duct wall. The edge rod cladding temperatures are significantly higher than those obtained for the unit cell model by the FASTRAN program. To account for these two-dimensional effects, a temperature margin must be defined and allowed in addition to the transient maximum temperature obtained by FASTRAN, as indicated in Section 5.1.2.1.

The detailed subchannel analysis code SCRIMP conducted a pseudo-steady-state analysis to define a typical value for the edge temperature margin. Section B.2 describes the SCRIMP program and the analysis method. Section B.2 presents the fuel assembly analysis results.

In the wire-wrap-spaced radial blanket assemblies, the maximum cladding temperature does not occur at the assembly edge due to the relatively large edge channel flow area. The cladding temperature of the blanket rod in the second row from the core is approximately equivalent to the maximum value occurring in the first row blanket rod, and the flow for the second row rod can be better modeled by a symmetrical unit cell. Based upon these approximate relationships, the maximum blanket cladding temperature condition is represented by the unit cell model of the second row rod in the FASTRAN transient analyses. Therefore, the edge temperature correction for the two-dimensional effect is not needed for the blanket cladding temperature.

Section B.4 presents the SCRIMP analysis used to confirm these blanket rod approximations.

B.2. SCRIMP PROGRAM DESCRIPTION

B.2.1. SCRIMP Code Description

The subchannel analyses were performed with the SCRIMP thermal-hydraulic computer code (Ref. B-1), adapted by the Swiss Federal Institute for Reactor Research (EIR) of Würenlingen, Switzerland, from the Aerojet HECTIC-II code (Ref. B-2). The program calculates steady-state coolant and surface temperatures for flow parallel to a rod bundle. It considers flow redistribution within the bundle, turbulent heat and momentum interchange, conduction between adjacent surfaces, and radiation from surface to surface. The code accepts axial and radial power distributions and correlations for friction factors and heat transfer coefficients for the different subchannel types (i.e., interior, wall, and corner) and includes the effect of grid spacers on bundle pressure drop. The set of differential equations for coolant temperatures is solved by applying a fourth-order Adams predictor-corrector scheme. The input specifies maximum allowable convergence errors for flow, coolant temperature, and surface temperature, and the code iterates between the flow and coolant temperature solutions until these limits are satisfied.

B.2.2. FASTRAN Validation Method

This section discusses the general approach of the validation analysis common to both the fuel and blanket assemblies. Sections B.3 and B.4 will present different models and data pertaining to the difference in fuel and blanket assembly cases.

SCRIMP differs from FASTRAN in that the former is a steady-state code, while the latter is a transient code. A steady-state code is employed, since an adequate transient-rod-bundle, gas-cooled, thermal-hydraulics code is not yet available. The drawback of a steady-state code is, of course, that it cannot account for the transient heat storage effect. To mitigate this discrepancy, the heat flux transferred from the FASTRAN modeled fuel pin into the unit cell coolant is converted into its equivalent pin power in the steady-state condition. This pin power conversion is performed at the transient peak cladding temperature. The converted pin power is then employed in the SCRIMP analysis as the power of that particular reference pin. The rest of the rod bundle fuel pins will have their respective powers derived from the decay heat distributions (Ref. B-4) superimposed on the power of the reference pin.

After reactor shutdown, gamma ray heating in the assembly steel components becomes a significant portion of the total assembly decay heat generation rate. This changes the distribution of assembly heat sources. The effect is most pronounced in peripheral subchannels where the relatively larger local heating rates, combined with the normal coolant flow diversion from peripheral to internal channels as flow rate decreases, cause significant edge channel undercooling during a design basis depressurization accident (DBDA) transient.

Figure 5-9 shows that the duct wall linear heat rate is ~4% of that in the assembly fuel rods immediately after shutdown, increasing to 5% in 3 h. Although not large on an absolute scale, all of the duct wall gamma heat is deposited directly in the peripheral subchannels. For a central fuel assembly 350 s after reactor trip, the duct wall contributes nearly 30% of the total heating in a wall subchannel and fully 50% of the total in a corner subchannel.

Duct wall linear heating rates were calculated by multiplying the value from Fig. 5-9 (the ratio of duct wall heating to fuel rod heating at time t) by the appropriate value from Fig. 5-7 (the ratio of linear heat rate of the fuel rods at time t to that immediately prior to shutdown), then multiplying by fuel rod linear power prior to shutdown:

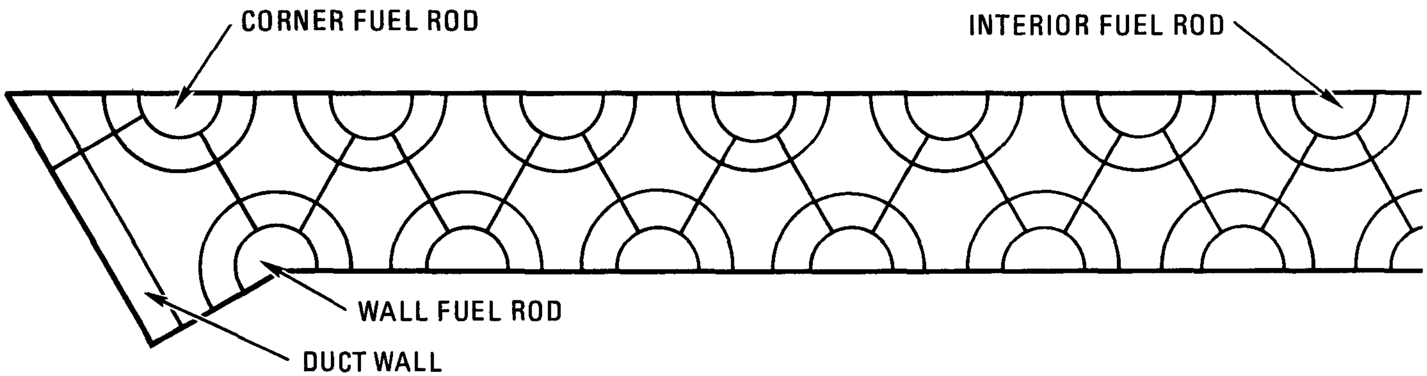
$$q'(duct,t) = \frac{q'(duct,t)}{q'(rods,t)} \times \frac{q'(rods,t)}{q'(rods,o)} \times q'(rods,o) \quad . \qquad \text{(B-1)}$$

Edge subchannel heat inputs were determined by weighing the duct and rod linear heat rates by the cross-sectional areas of the adjacent wall and fuel rods.

The SCRIMP code models a six-axial section fuel pin to simulate the six axial nodes modeled by FASTRAN. SCRIMP allows radial pressure equalization among all bundle flow channels at the inlet of each axial section. Hence, a limited amount of enthalpy due to this flow redistribution is exchanged. However, for conservatism, turbulent mixing is not permitted. The correct amount of coolant flow for SCRIMP calculations can be determined by parametric calculations, matching the calculated core pressure drop (from lower to upper plenum) against the FASTRAN results at the transient peak cladding temperature.

Rod bundle strip modeling is employed. A full-assembly thermal-hydraulic calculation would be extremely costly and is deemed unnecessary. Assuming symmetry, only half a strip model is needed, as shown in Figs. B-1 and B-2. The figures show the full length of a strip model. For fuel assembly analysis, partial model length is sufficient when radial symmetry is also assumed. However, a full-length model is needed for the blanket assembly analysis, since the blanket assembly has a strong radial power asymmetry.

Although these validation calculations were for a different GCFR design, the general approach should sufficiently validate the code.



FUEL ROD O.D. = 7.46 MM
FUEL ROD PITCH = 10.4 MM
EDGE SPACING = 82%

CLADDING THICKNESS = 0.51 MM
DUCT WALL THICKNESS = 3.9 MM

Fig. B-1. SCRIMP fuel assembly model

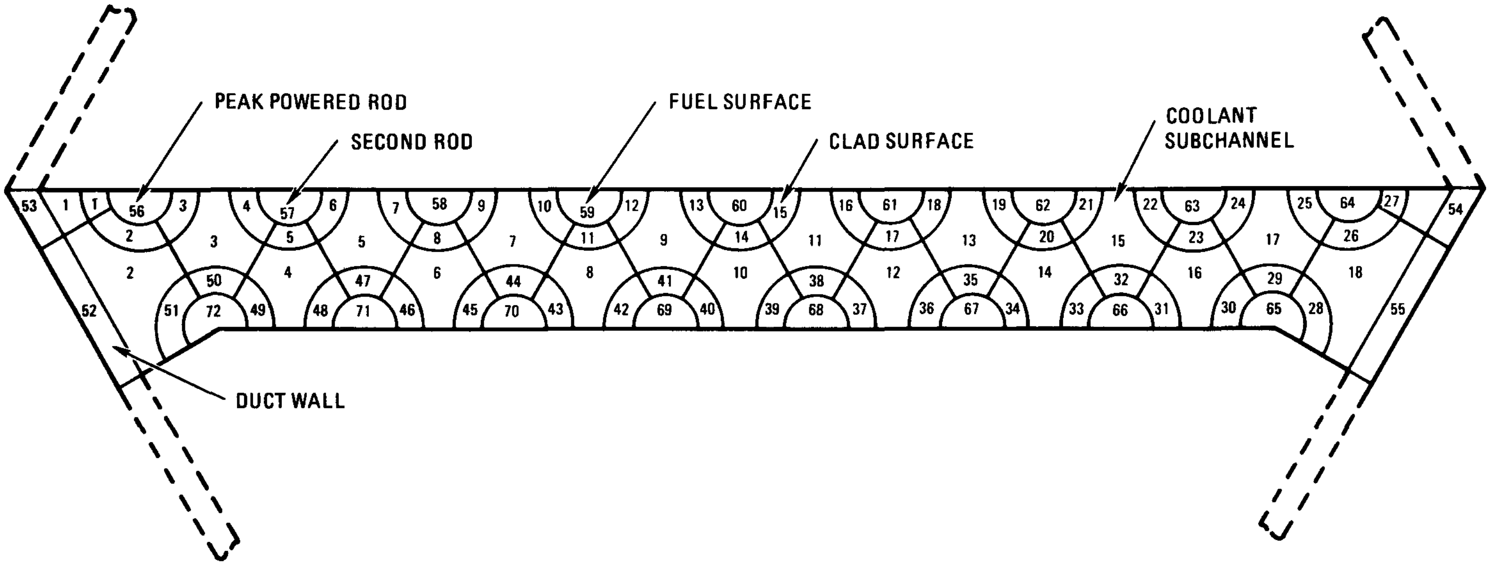


Fig. B-2. SCRIMP blanket assembly model

B.3. FUEL ASSEMBLY ANALYSIS

B.3.1. Analysis

In the FASTRAN analysis of the DBDA transient, the nominal fuel rod cladding temperatures were found to occur at 354 s into the transient. Conditions at that time were input to a SCRIMP fuel assembly model, and steady-state temperature profiles were calculated. To avoid problems associated with transient heat storage in the rods, surface heat flux levels, rather than power generation rates, calculated an equivalent power for input to SCRIMP.

The fuel assembly configuration used in this analysis was a modification of the Ref. B-3 design. While the rod o.d. of 7.46 mm and rod pitch of 10.4 mm were retained, the spacing between the edge rods and the duct wall was increased to 2.4 mm, and the grid spacer hanger rods were moved from the assembly corners to the interior to more accurately simulate current design trends.

Figure B-1 shows a SCRIMP model schematic. The model was sized so that the ratio of interior channel to peripheral channel flow areas is similar to that of the full bundle. The assembly analyzed was assumed to have no radial power gradient. Figure B-3 gives the axial power profile. Heat transfer by conduction was allowed between contiguous surfaces, and radiation heat transfer viewfactors were determined and included in the model.

B.3.2. Results

Figure B-4 shows the temperatures in the peripheral rods at the axial location corresponding to peak surface temperatures calculated by SCRIMP. For the particular fuel assembly geometry and the hydraulic conditions analyzed, the peripheral subchannels are undercooled at this axial location, a situation opposite to that at full power. This means that the peripheral channels run hotter than those in the interior. For this particular case,

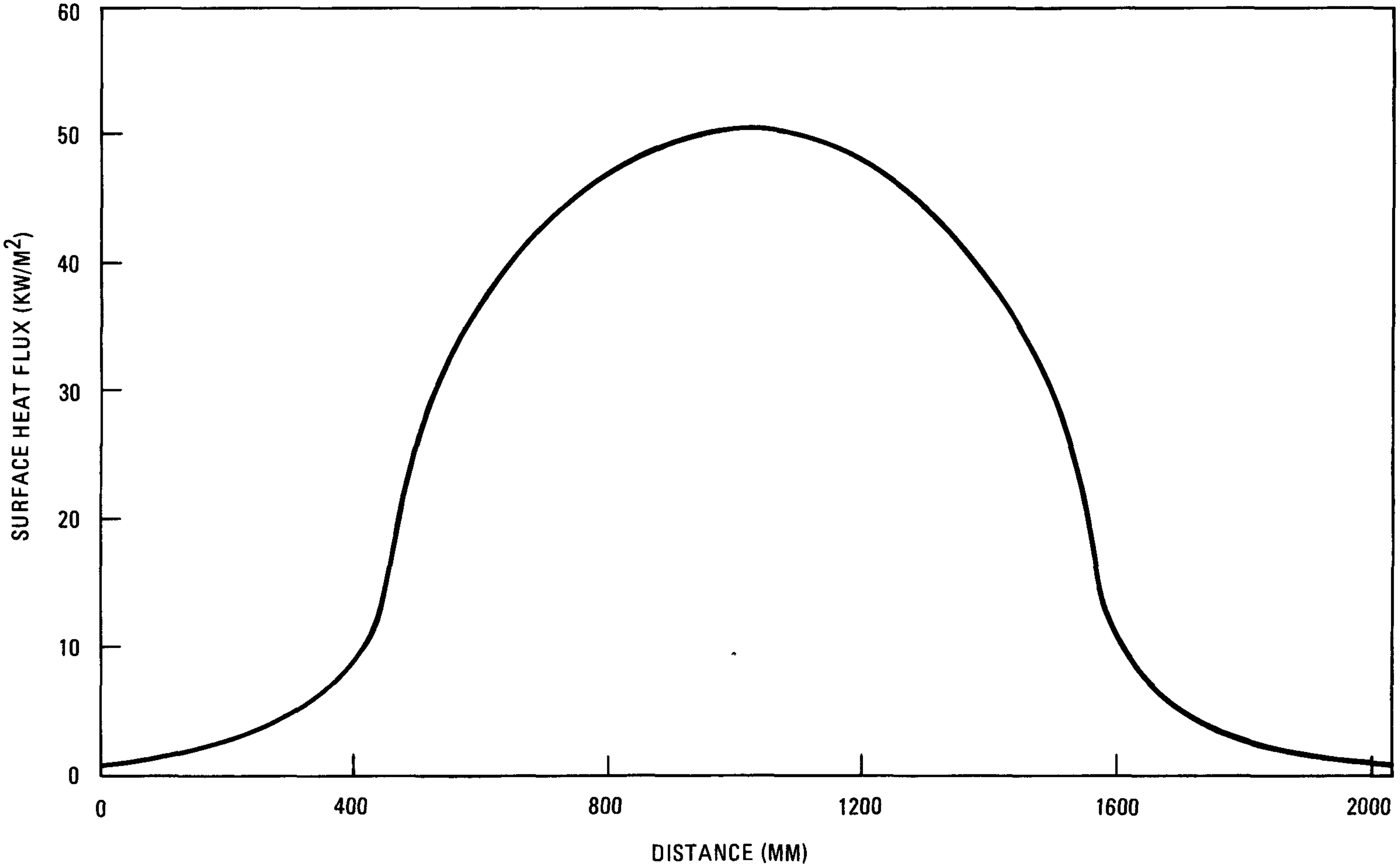


Fig. B-3. Rod distributed surface heat flux

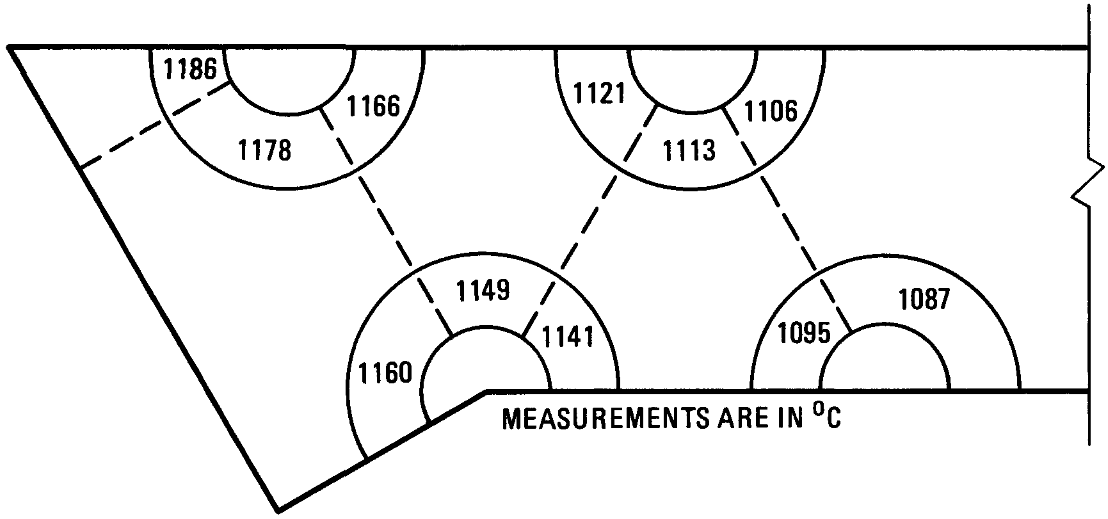


Fig. B-4. Peripheral rod temperature at axial location corresponding to peak temperature the peak cladding temperature occurs on the outer face of the corner rod and is 122°C (252°F) higher than surface temperatures of interior rods. At these power conditions, the temperature drop through the cladding is on the order of 2°C (3.6°F), so that rod surface and clad midwall temperatures are essentially the same.

Both absolute temperature levels and lateral temperature gradients across the fuel rods are important to rod bowing. For this configuration, the gradient across the corner rod is similar to that across the wall rod, and both are moderate compared to full-power gradients. Circumferential conduction in the rod cladding moderates cladding gradients at low power more strongly due to the increased resistance to convective heat transfer.

B.3.3. Discussion

The peak interior fuel rod temperatures calculated by SCRIMP are within 10°C (18°F) those calculated by FASTRAN with its unit cell representation. This analysis has shown that the temperature gradient between the peripheral and interior rods is on the order of 125°C (257°F) for the configuration analyzed, thus confirming that a 100°C (212°F) tmeperature margin should be allowed in addition to the maximum fuel cladding temperature obtained by FASTRAN one-dimensional analysis, as mentioned in Section 5.1.2. The temperature margin is a strong function of many parameters, including rod diameter and pitch, edge spacing, coolant flow regime, degree of turbulent mixing between adjacent subchannels, amount of power generation in the duct wall due to gamma heating, emissivity of radiating surfaces, etc. For example, reducing the edge spacing from 2.4 to 1.5 mm causes the peripheral/interior temperature gradient to double. Decreasing the surface emissivity from 0.6 to 0.3 causes the gradient to increase by 25%. All of these factors are being considered in the redesign of the fuel assemblies to achieve as low a gradient as is consistent with efficient operation at full power and to meet the plant condition (PC)-5 temperature limit of 1300°C (2372°F).

B.4. PEAK POWERED RADIAL BLANKET ASSEMBLY ANALYSIS

B.4.1. Analysis

The steady-state rod bundle code SCRIMP calculated and validated the DBDA radial blanket peak cladding temperature obtained with the transient system code FASTRAN. A full strip model was adapted to model the rod bundle thermal-hydraulic effects of a maximum powered radial blanket assembly adjacent to the active core. Figure B-2 depicts the model. A full model, rather than a strip model, was employed, because the radial power distribution in a blanket assembly exponentially decreases outward. Therefore, a full strip model is needed to represent adequate rod bundle effects, such as flow redistribution.

The decay power in a radial blanket assembly consists of the decay power generated by the blanket assembly itself and net gamma-ray transported from the active core to the blanket assembly. The decay power distribution employed the latest algorithm developed in Ref. B-4, similar to the form presented in Section 5.1.4.1, and the gamma transport effect in Ref. B-5. The peak-powered rod is the corner rod next to the active core (rod No. 56 in Fig. B-2). Since FASTRAN can only model a single rod with a unit cell flow channel surrounding it, the transient study modeled the second highest powered rod (rod No. 57 in Fig. B-2). Due to the rod bundle heat transfer effects and the large edge channel of the highest powered rod, a unit cell modeling of the second highest powered rod is believed to generate equivalent hot spot cladding temperature that usually occurs on the peak powered rod.

As mentioned in Section B.1, to mitigate the difference between a transient and a steady-state analysis, the unit cell heat flux of the second rod output from FASTRAN was adapted as the basis for the second rod power input to the SCRIMP code. The decay power radial distribution was then superimposed on the second rod power to obtain the rod power for the rest of the strip model rods. Table B-1 lists the resulting power distribution at

TABLE B-1
LINEAR POWER DISTRIBUTION IN THE MAXIMUM POWERED
RADIAL BLANKET ASSEMBLY

	Linear Power (W/cm)			
	At full power		At 1641 s of DBDA	
	Maximum	Average	Maximum	Average
Fuel rod No.[a]				
56	452.28	218.39	11.094	5.748
57	365.07	176.28	6.736	3.490
58	308.75	149.08	5.223	2.706
59	259.81	125.45	4.350	2.254
60	226.67	109.45	3.846	1.993
61	195.09	94.20	3.362	1.742
62	171.47	82.80	3.013	1.561
63	151.79	73.29	2.704	1.401
64	133.41	64.62	2.704	1.401
72	415.29	200.53	11.547	5.983
71	365.07	176.28	6.736	3.490
70	308.75	149.08	5.223	2.706
69	259.81	125.45	4.350	2.254
68	226.67	109.45	3.846	1.993
67	195.09	94.20	3.362	1.742
66	171.47	82.80	3.013	1.561
65	151.79	73.29	3.154	1.634
Duct wall surface[a]				
52	28.17	13.60	2.531	1.311
53	11.78	5.69	1.058	0.548

[a]Fuel rod and duct wall surface numbers are based on Fig. B-2.

the transient peak cladding temperature at 1641 s, predicted by FASTRAN. Figure B-5 shows the corresponding axial power distribution output from FASTRAN.

Reference B-3 gives the radial blanket rod cladding i.d. as 1.865 cm, the o.d. as 1.965 mm, and a lattice pitch as 2.105 cm. The rod bundle is wire wrapped. Blanket fuel length is 203 cm.

The model flow rate was determined by benchmarking the calculated core pressure drop (from lower to upper plenum) against the FASTRAN 5.103 kPa (0.74 psi) by parametric calculations. The hot spot factors employed for calculating the hot spot cladding surface temperature are identical to the conservative ones presented in Table 5-7.

B.4.2. Results

The nominal maximum clad surface temperature of 1046°C (1915°F) compares favorably with the FASTRAN 1075°C (1967°F). This peak cladding temperature occurred on the side of the peak powered rod away from the active core. Figure B-6 illustrates the axial temperature profiles of this cladding surface and its associated coolant channel. The radial temperature gradient across the cladding is small, only a few degrees. Figure B-7 depicts the cladding surface temperatures of the rods adjacent to the peak-powered rod at the elevation of the hot spot temperature. As shown in Fig. B-7, the difference of the maximum clad surface temperature between the first (corner) and the second rod is only ~25°C (55°F).

B.4.3. Discussion

This SCRIMP code analysis was performed for the outdated 300-MW(e) GCFR demonstration reactor of Ref. B-3. However, several general remarks can be learned from this study.

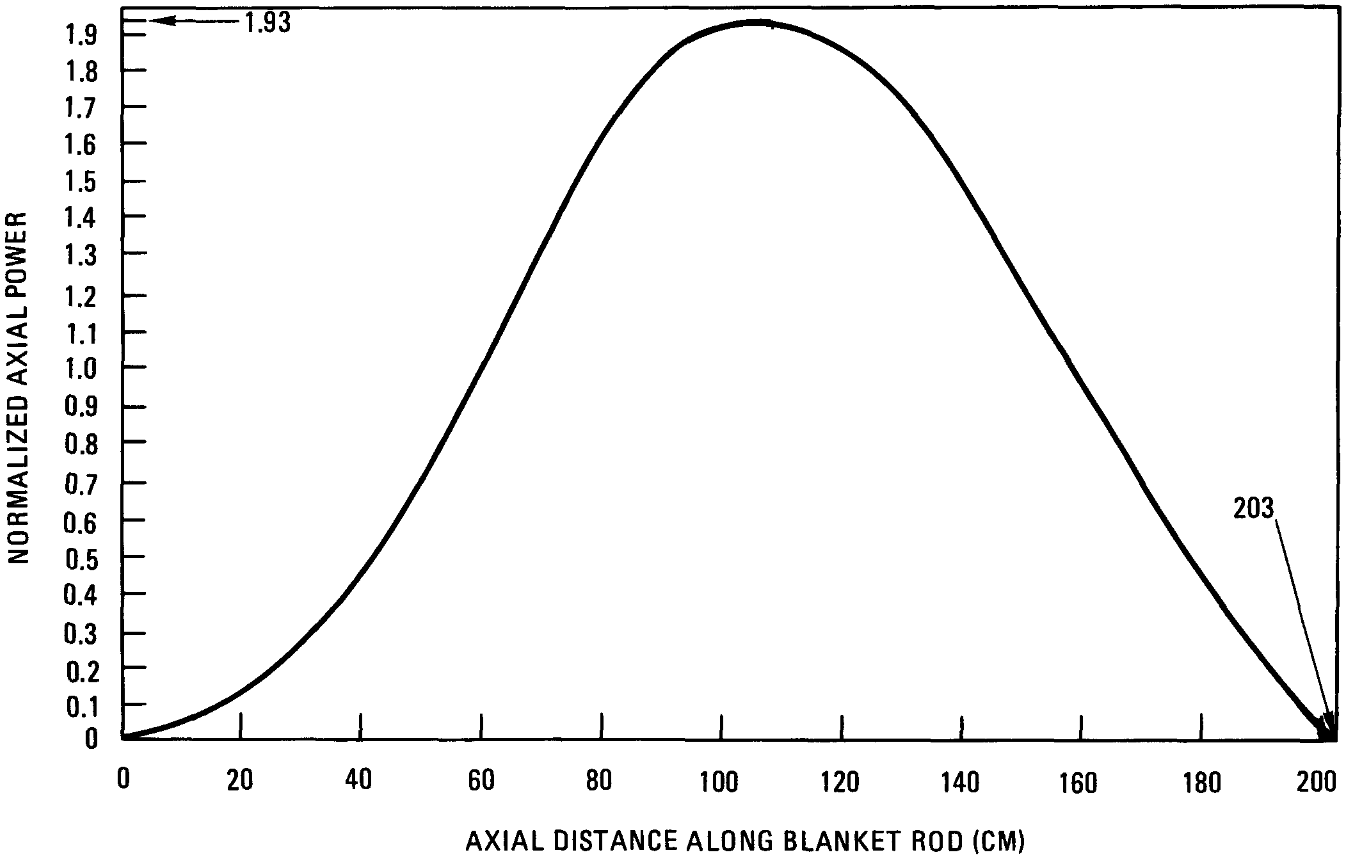


Fig. B-5. Normalized axial power distribution of radial blanket rod 1641 s for 4.84-cm^2 (75-in.^{2}) DBDA

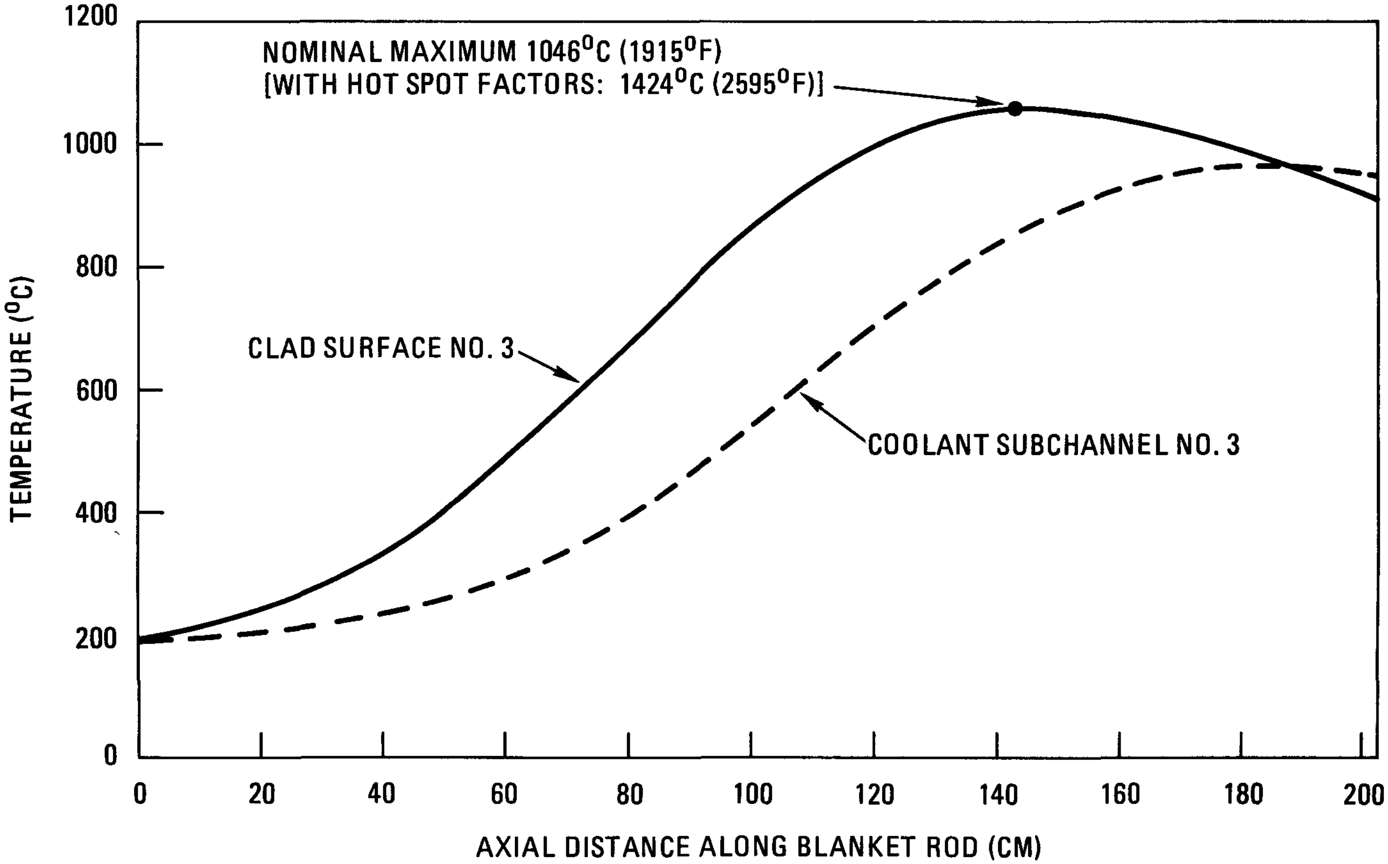


Fig. B-6. Nominal maximum axial temperature profiles of clad surface No. 3 and coolant subchannel No. 3

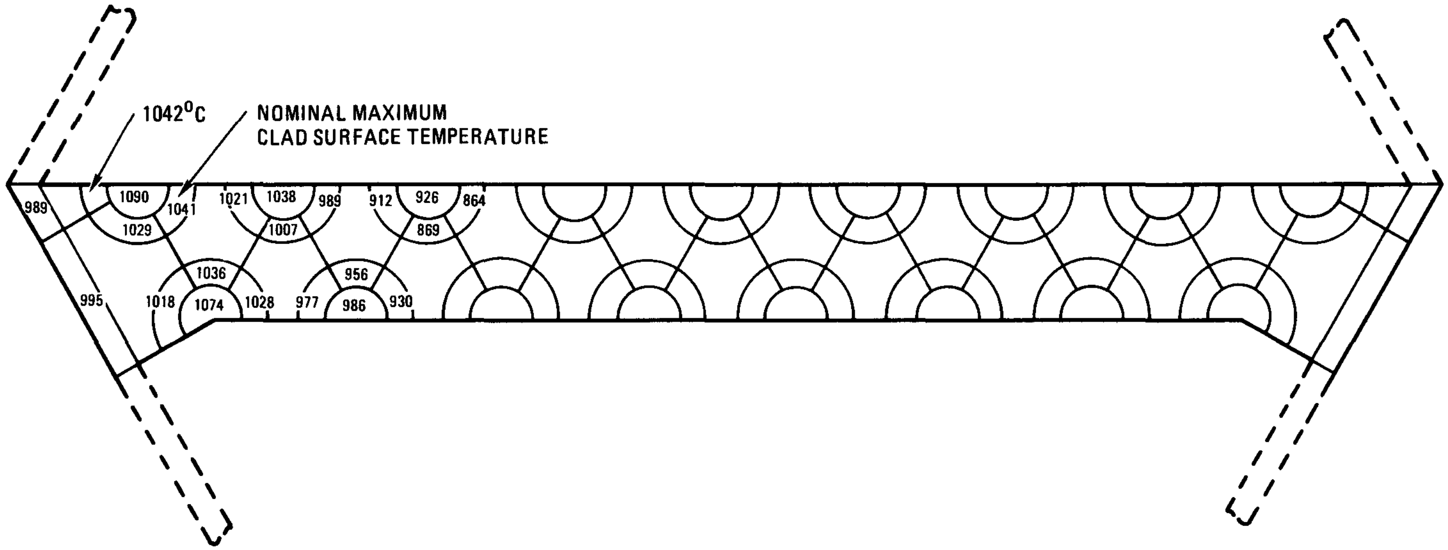


Fig. B-7. Radial temperature distribution at the axial location of the maximum clad temperature

The maximum clad surface temperatures calculated by SCRIMP and FASTRAN compare favorably. The SCRIMP code rod bundle study showed that the maximum clad surface temperatures of the peak-powered (corner) rod and the second rod are comparable. Thus, FASTRAN is adequate to model the unit cell of the second highest powered blanket rod.

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