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CLINCH RIVER

BREEDER REACTOR PROJECT

50-537

PRELIMINARY SAFETY ANALYSIS REPORT

VOLUME 9

PROJECT MANAGEMENT CORPORATION

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C. Estimated Reaction and Response Times

1. The time required for the initial accident assessment of the most serious design basis accident may require 15 minutes. This time is an estimate based on the operation of the reactor instrumentation used to follow the course of accidents. Based on TVA's experience, the time required to perform an initial dose projection and notify offsite authorities can be accomplished in 15 minutes.

For the most serious design basis accident, the projected two-hour doses at the exclusion area boundary do not reach the protective action guide level for evacuation.

2. The time required to warn all resident and transient persons in any evacuation sector will conform to the requirements of 10 CFR 50, Appendix E-1982.

3. The estimated elapsed time, after the initial warning, to evacuate the 2-mile emergency planning zone (EPZ) is 4 hours. The estimated evacuation time for the 5-mile EPZ is 6 hours, 35 minutes. The estimated evacuation time of the 10-mile EPZ is 8 hours 45 minutes. Each estimate contains a 1-hour 50-minute preparation time factor.

4. These evacuation time estimates were prepared by the Traffic Management Division of the Tennessee Department of Transportation.

   a. Figures 13.3A-5 and 13.3A-6 are maps showing all roads within 10 miles of the Clinch River Project. Also indicated are the 2-, 5-, and 10-mile EPZ.

   b. Table 13.3A-1 shows the transient and resident populations in the 16 directional sectors within 10 miles of the Clinch River Project. This table uses 1980 census data.

   c. Table 13.3A-2 shows the estimated transient and resident populations in the 16 directional sectors within 10 miles of the Clinch River Project. This table uses the projected population figures for year 2020. The projected population figures come from a report prepared by the Firm of Dames and Moore dated June 16, 1981.

   d. Private automobiles will be the primary means for evacuating the population. Buses are expected to be used to evacuate schools and other institutions. This procedure will be specifically addressed in the CRBRP-REP.

5. Table 13.3-1 gives the agencies involved in the CRBRP emergency plan.

TABLE 13.3A-1
MAXIMUM RESIDENT AND TRANSIENT
POPULATION DISTRIBUTION WITHIN
10 MILES OF THE DEMONSTRATION PLANT
FOR CENSUS YEAR 1980

Sector Designation	Radial Interval (miles) 0-1	1-2	2-3	3-4	4-5	5-10
N	0	0	14	184	0	2,000
NNE	0	0	0	0	22	4,400
NE	0	0	0	8	80	7,191
ENE	10	10	0	8	0	4,728
E	20	30	50	398	3,568	5,172
ESE	20	30	50	187	159	2,300
SE	0	59	50	460	110	7,200
SSE	0	300	79	90	320	2,000
S	0	89	50	120	160	1,120
SSW	10	69	50	80	90	936
SW	20	80	119	110	140	1,292
WSW	20	70	80	193	340	5,000
W	0	130	114	110	991	6,764
WNW	10	94	170	10	60	4,676
NW	30	44	0	10	40	3,972
NNW	10	514	316	850	120	1,100
Sum for Radial Interval	150	1,519	1,142	2,818	6,200	59,851
Accumulative Total up to Radius Indicated	150	1,669	2,811	5,629	11,827	71,680

CLINCH RIVER

BREEDER REACTOR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 14

INITIAL TESTS AND OPERATION

PROJECT MANAGEMENT CORPORATION

CHAPTER 14 INITIAL TESTS AND OPERATION

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CHAPTER 14.0 INITIAL TESTS AND OPERATION

This chapter provides information relating to the initial plant startup and operation program to show that the licensee plans to develop and conduct a comprehensive test program on this first of a kind plant, and that necessary early planning has been done for successful achievement of these goals.

The need is recognized for development of a comprehensive preoperational and initial startup test program for the CRBRP plant, the preparation of adequate test instructions for carrying out the programs, the proper conduct of the test programs, and assuring the validity of the test results. The test programs will provide additional assurance that the plant has been properly designed and constructed and is ready to operate in a manner that will not endanger the health and safety of the public; that the operating instructions for operating the plant safely have been evaluated and demonstrated; and that the plant operations personnel are knowledgeable about the plant procedures and operating instructions and fully prepared to operate the plant in a safe manner.

The test programs will also include testing for interactions such as the performance of interlock circuits in the reactor protection systems. It will be determined that proper permissive and prohibit functions are performed and that circuits normally active and supposedly unaffected by the position of the mode switch perform their function in each mode. Care will be taken to ensure that redundant channels of equipment are tested independently.

The test program development will utilize and incorporate preoperational and startup test experience from LWR's and FFTF. PSAR, Appendix H, paragraph I.C-5, "Procedures for Feedback of Operating, Design, and Construction Experience," provides additional information on the procedures for ensuring such experience utilization.

14.1 DESCRIPTION OF TEST PROGRAMS

The initial test program for the plant is divided into two parts: preoperational testing, and the initial startup testing.

Phases 1 and 2 preoperational tests are those conducted prior to fuel loading to demonstrate the capability of structures, systems, and components to meet performance requirements, including safety related requirements. These tests are used to demonstrate that overall plant performance is acceptable and that the CRBRP is ready for initial installation of fuel. For scheduling purposes, the preoperational tests are divided into two phases. Phase 1 is defined as testing following plant turnover from the constructor to initial introduction of sodium into the Heat Transport System (HTS). Phase 2 testing is defined as the plant testing which requires sodium in the HTS prior to initial core load.

Startup testing consists of such activities as fuel loading, precritical tests, low power tests (including critical tests), and power-ascension tests performed after fuel loading to completion of acceptance testing that confirm the design bases and demonstrate, where practical, that the plant is capable of withstanding the anticipated transients and postulated accidents.

Startup testing is also divided into two phases for scheduling purposes. Phase 3 is defined as the testing period beginning with initial core load and extending to 5% power. Phase 4 is defined as the power ascension test period
61 and covers the power testing from 5% to 100% power.

14.1.1 Preoperational Test Program

The objectives of the preoperational test program are to demonstrate the capability of structures, systems, and components to meet performance requirements; to assure that, to the extent possible, procedures for operating the plant have been used and evaluated; and that the operating organization acquires sufficient knowledge about the plant features and procedures to operate the plant in a safe manner. The preoperational test program will demonstrate not only that the design of systems, structures and components meet the objectives, but that construction of the plant has been done in a manner that assures that the plant can be operated safely.

The preoperational test program will begin only after a very significant portion of the plant construction is complete. Before a structure, system, or component is preoperationally tested, activities on it must be essentially complete, with those incomplete portions clearly documented in the test report.

Before preoperational tests are started on a system, structure, or component, construction tests such as system flushing and cleaning, wiring checks, and leak tightness tests must be completed to the extent that meaningful test results are obtained. Each system, subsystem, or component will have successfully passed the construction test and gone through a turnover procedure prior to commencement of preoperational testing. In addition, initial calibration of instrumentation, and subsystem component functional
61 tests must be completed prior to subsystem preoperational testing.

14.1.2 Startup Test Program

The objectives of the startup test program are to assure orderly, safe fuel
61 loading, low power testing and approach to full power testing; and to confirm the design bases and demonstrate, where practical, that the plant is capable of withstanding the anticipated transients and postulated accidents.

The startup test program begins with fuel loading and ends with the of the Project Office accepting the CRBRP from the contractors based on satisfactory
61 evaluation of test results and correction or acceptable provision for corrections of all identified deficiencies and incomplete items. This program is started only after conclusion of all preoperational tests that can be performed with no fuel in the reactor.

61 The startup test program is composed of the following activities:

61 Fuel Loading Tests (Scheduled in Phase 3) - provide assurance of a safe, orderly loading of the core, taking into consideration the first-of-a-kind core, the available nuclear instrumentation, and reactor control.

61 Precritical Testing (Scheduled in Phase 3) - includes those tests from initiation of core loading to criticality. These tests will assure that the startup proceeds in a slow and orderly manner, that changes in reactivity will be continuously monitored, that operations personnel are aware of core reactivity state, and all systems are aligned and in proper operation.

61 Low Power Testing (Scheduled in Phase 3) - includes those tests between criticality and a 5 percent power level. These tests will confirm nuclear design parameters before nuclear heating and give confidence that the reactor power can be increased. Nuclear instrumentation will be confirmed at increased power levels.

61 Power Ascension Testing (Scheduled in Phase 4) - includes those tests at various power levels between 5 and 100 percent power. Confirmation of reactor and plant design parameters are obtained at progressively higher power levels, each step giving confidence that the next higher power level can be safely accommodated.

14.1.3 Administration of Test Program

This description of the administration of the test program applies to both the preoperational and startup phases. In planning and carrying out this program, the guidelines of Regulatory Guide 1.68 will be used insofar as they apply to the LMFBR. This includes all of the Guide except those portions of Appendices A&C that are unique to light water reactors. Other regulatory guides will be reviewed at the time detailed test instructions are being developed to establish which guides have applicability to the program.

14.1.3.1 RESPONSIBILITIES

61 The Project Office (DOE) has overall responsibility for the plant initial test program. Portions of the program have been assigned to others as follows.

TVA is assigned responsibility for conduct of the Preoperational and Startup Testing along with its responsibility as plant operator. It also has responsibility for review and approval of test specifications for preparation of test instructions from the test specifications, for evaluation of adequacy
61 of operating and emergency instructions during the test program, for on-site approval of test results, and for recommending plant modifications as a result of deficiencies discovered during testing. As described in Section 14.2, the TVA normal operating organization will be augmented during this test period. The responsibility for performance of preoperational tests will be assigned by

the plant manager to a TVA Preoperational Test Section. The responsibility for performance of startup tests will be assigned to the plant technical staff augmented by technical specialists from TVA's central office.

The Project Office has responsibility for review and approval of all aspects of the test program including scope, content, schedule, test specifications, test instructions, test results, and any plant modifications required as a
61 result of the test program.

Westinghouse ARD as Lead Reactor Manufacturer (W-LRM) is assigned responsibility for preparation of technical aspects of the initial plant test program. In carrying out this assignment they will utilize the services of the Reactor Manufacturers, W-ARD as Reactor Manufacturer (W-LRM), General Electric (GE-RM), and Atomics International (AI-RM) and coordinate with and utilize the input of Burns and Roe as Architect Engineer (A-E). This includes early planning of scope, schedule, and sequencing of the testing interfacing the construction schedule. W-LRM is responsible for preparation of test specifications, and for reviewing the test instructions and test results for NSS systems under their cognizance. W-LRM is responsible for design of plant modifications required to their systems as a result of deficiencies discovered
61 during the test program. W-LRM will assign on-site personnel for technical direction during the test program.

Burns and Roe as Architect Engineer (A-E) is assigned responsibility for preparing test specifications, and for reviewing for technical adequacy the test instructions and test results of those BOP and NSS systems under their
61 congnizance. The A-E is also responsible for design of any plant modifications required to their systems as a result of deficiencies discovered during the test program. The A-E will assign on-site personnel for technical direction of tests to their systems.

61 W-LRM and the A-E will establish on-site staff during the test and startup period for technical direction of the initial test and startup program. This technical direction will include supplying technical advice and information to operations personnel to assist them in making decisions. This staff will have the capability to support the TVA plant staff in both operations and testing. Technical direction does not include supervision of operations personnel. The responsibility for safe operation of the plant rests with TVA as the plant operator.

61 Stone and Webster Engineering Corporation as the Constructor is assigned responsibility to assist TVA as plant operator in assuring that all prerequisites are met before tests are started, for insertion of detailed scheduling of the test program into the plant construction schedule, and for assisting as required in repair or modification of the plant as a result of deficiencies found in the test program.

61

14.1.3.2 <u>Procedure Outline Preparation</u>

The Project Office has assigned the responsibility for preparation of test specifications, operating, maintenance, and surveillance procedure outlines to Westinghouse as Lead Reactor Manufacturer and Burns and Roe as Architect-Engineer. Using information from the A-E, GE, and AI, W-LRM will provide a current Test Sequence document that describes the initial test program content and schedule. This document will be reviewed,and approved by The Project Office as a basis for further detailed planning of the initial test program. A Test Abstract document may be prepared that provides a short description of each test and gives the objectives for the test. A Test Network will be developed that shows the schedule for each test. These documents will be reviewed and approved by the Project Office.

Using this planning information as a basis, W-LRM and the A-E will prepare
61 test specifications that describe each test requirement in considerable
detail. These specifications will describe prerequisites, test objectives,
general test methods, and acceptance criteria. Test specifications will be
61 reviewed and approved by the Project Office.

Using test specifications as a basis, TVA will prepare detailed test
instructions. Format and content of these test instructions will conform to
the guidance given in Appendix C of Regulatory Guide 1.68. Operating
instructions will be available at the time these test instructions are
prepared, so that the operating instructions may be referenced in aligning
systems for testing. These detailed test instructions will be reviewed and
approved by the Project Office and TVA.
Final review and approval for use will be performed by TVA as the plant
61 operator.

61 Changes to test instructions that modify the objectives, intent, or
significantly change the method of test performance will receive the same
review as the initial instructions. Minor changes that do not modify the
objectives, intent or significantly change the method of test performance may
be made by the testing organization and will be documented and subsequently
reviewed as previously described.

61 Each test instruction will contain a prerequisites section that will describe
in detail all prerequisites, including construction related, that must be
satisfied before a test is performed. Sign-off sheets will be provided with
each instruction to record verification that prerequisites are satisfied.
Completion of this sign-off sheet will be mandatory before starting the test.

14.1.3.3 Conduct of Tests

TVA as plant operator has been assigned responsibility for conduct of
Preoperational and Startup Tests. This includes review and approval for use
of all test specifications, preparation of test instructions from the test
specifications, performing all operator manipulations required under the plant
operating license, assuring that all test prerequisites are satisfied,
performance of details of the instructions, and collection of data for
approval of test results. TVA will be assisted in the performance of this
61 responsibility by W-LRM and the A-E and any necessary subcontractors acting as
technical directors for tests under their cognizance.

During the conduct of the tests, plant operating and emergency instructions will be tested wherever possible. This will help assure that the instructions can be used to safely operate the plant, and provides further assurance that the operator is familiar with the instructions and thoroughly trained to operate the plant.

Test results will be compared to acceptance criteria by the on-site test group. Deficiencies will be immediately reported to the Project Office,
61 W-LRM, and/or the A-E. Each deficiency will be evaluated by these participants, and the appropriate corrective action specified, such as retesting or instruction change. These corrective actions will be reviewed
61 and approved by the Project Office and TVA as plant operator.

61 A detailed review of test results will be made by W-LRM or the A-E for tests within their scope of responsibility. This detailed review will confirm the technical adequacy of the system, component, or structure to operate in accordance with design specifications.

14.1.3.4 Test Program Schedule

Figure 14.1-1 shows the schedule for each major phase of the initial test program. It also shows the schedule for preparation of plant instructions, key milestones in staffing for operation, and augmentation of the plant staff for initial startup test assistance.

As shown on this schedule, all plant instructions will be prepared before fuel loading. Operating, maintenance, and surveillance test instructions will be started at about the same time as initial test instruction preparation, and is
61 scheduled for completion before fuel loading.

A small group of key TVA operations personnel will be on-site about three years before start of the preoperational test program. From this nucleus the on-site operations staff will increase in size at a sufficient rate to provide adequate support for the preoperational test program. This schedule allows sufficient time for plant familiarization, and procedure review before testing starts. As described in Section 13.2 and shown in Figure 13.2-1, basic nuclear courses for operators and specialists training for technical personnel as well as assignment to a sodium cooled fast reactor will have preceeded this period.

14.1.4 TEST OBJECTIVES OF FIRST-OF-A-KIND PRINCIPAL DESIGN FEATURES

The following Test Abstracts are provided per US-NRC NUREG-75/087 - Section 14.1, Review Responsibilities item 2 for special, unique or First-of-a-Kind principal design features included in the CRBRP.

14.1.4.1 IN-VESSEL TRANSFER MACHINE

The only equipment of the reactor refueling system, which is considered first-of-a-kind and unique to the CRBRP, is the in-vessel transfer machine (IVTM). The IVTM is installed in the reactor head during reactor refueling and is discussed in detail in Section 9.1.4.4.

In order to minimize preoperational testing of reactor refueling system equipment at the CRBRP, the IVTM will be tested and checked out extensively at the off-site test facility (currently planned at ETEC).

The off-site tests are scheduled early in the program to ensure corrective actions can be taken to qualify the IVTM for CRBRP service without jeopardizing the overall plant construction schedule should any IVTM deficiencies be uncovered.

The IVTM prototype will be tested extensively to demonstrate that the IVTM meets its specification performance and design requirements. The complete and integrated IVTM assembly will be tested, including the control console with the minicomputer.

After the IVTM has been assembled at the test site, and the assembly has been checked out, the IVTM will first be subjected to individual and integrated checkout tests. Following this, the IVTM will be performance tested simulating core assembly transfers.

The tests will be performed in special test facilities containing a cluster of at least seven simulated core assemblies. The cluster will be capable of relative vertical and horizontal displacements and side loads.

A. INDIVIDUAL CHECKOUT TESTS

The purpose of the individual checkout tests is to verify that the following IVTM functions can be performed:

61 | 1) Grapple and release of a Core Special Assembly (CSA).

61 | 2) Raise and lower a CSA to positions corresponding to those encountered in the reactor vessel.

61 | 3) Identify and orient a CSA.

61 | 4) Provide adjacent CSA holddown when removing a CSA from the CSA cluster.

5) Provide cover gas containment and seal leakage detection capability.

Specific tests will include the following:

1) Calibration and checkout of all IVTM interlocks, load cells, and the entire load control system.

2) Verification of all functions of the core assembly identification system.

3) Checkout of the grapple and holddown sleeve drive systems including removal of an artificially jammed core special assembly.

4) Calibration and checkout of the grapple and holddown sleeve position indication systems.

5) Verification of the seal leakage monitoring and the seal pressurization control systems.

B. INTEGRATED CHECKOUT TESTS

The purpose of these tests is to prove that the IVTM meets the following objectives:

1) The IVTM can perform the sequence of functions listed in Section A which are required to transfer a core assembly in accordance with given operating profiles when using computer and manual controls.

2) Insertion and removal of core assembly into and from the core can be accomplished under maximum misalignment in combination with maximum core assembly push and pull loads.

61 3) Release of core assembly into an incorrect core position is prevented.

4) Release of a core assembly into a transfer position in the absence of a core component pot cannot be accomplished.

5) Premature release of a core assembly during operation over the core is prevented when the core assembly is at a vertical position higher than a small tolerance above the fully seated position.

C. PERFORMANCE TESTS

These tests are designed to simulate reactor refueling operations equivalent to at least five refueling periods.

61 The tests will be performed with a cluster of seven core special assemblies. The core special assembly cluster will be offset in relation to the IVTM to simulate core assembly insertion and removal under misaligned conditions. Integrated operations of the IVTM, control console, and computer will perform simulated actual refueling operations. The major test objective is to demonstrate that all IVTM components, especially dynamic seals, will perform for a minimum of one refueling cycle. The test goal for all mechanical components of the IVTM (excluding elastomeric seals) is to demonstrate operation without failure. Post-test inspection of the mechanical components will establish the acceptability of component wear.

The following results will be obtained from these tests:

1) Wear data of dynamic seals.

2) Wear data of mechanical components.

3) Establish transfer cycle speeds for automatic and manual operation.

4) Wear data of core assembly identification pawl.

5) Any operational limitations.

6) Any deficiencies in the operating components and/or in the design.

7) Verify the computer control of the fuel transfer cycle.

8) Verification of the core assembly identification system with respect to wear data obtained in item 4 above.

9) Verification of checkout and operational procedures.

D. PREOPERATIONAL IVTM TESTS AT CRBRP

Those IVTM operations which are not simulated in the special test facilities will be performed after IVTM installation, adjustments, and checkout at the CRBRP reactor small rotating plug prior to fuel loading. These tests will include:

61 | 1) Insertion and removal of core special assemblies into and out of a Core Component Pot (CCP), and transfer of those assemblies between selected core addresses.

2) Integrated operational tests of the IVTM with the reactor rotating plugs (RRP).

3) Integrated tests of the IVTM to demonstrate design protection against off-normal operations to confirm accident analysis assumptions.

4) Integrated operational tests of the IVTM and interfacing reactor refueling system equipment to assure joint operability.

Before the IVTM is installed on the small rotating plug for the first time, and after that, each year before refueling, all IVTM functions required for transfer will be checked out in the dry IVTM maintenance and storage facility.

61 | 14.1.4.2 PRIMARY/SECONDARY SODIUM PUMP

Prototype Pump Tests

A. Prototype Pump Water Tests at Supplier's Facility

The objectives of water testing is to make final trim to the impeller and verify that the hydraulic performance of the pump meets the specification requirements regarding head and flow relationship, Net Positive Suction Head requirements, and to verify coast down head of flow versus time, bubbler performance (level control), capability to operate for a sustained endurance period, and operation at a loop impedance comparable to 2 loop plant operations (Runout to 41,000 gpm).

Functions which will be tested are:

1. Heat versus flow for plant loop impedance with speed as the variable.

2. Head versus flow for constant speed with variable impedance for several different speeds. Check for hydraulic instabilities as reflected in slope of H-Q curves.

3. Net Positive Suction requirements will be checked by reducing cover gas pressure while operating at rated speed and head and flow until degradation of performance or excessive vibration is detected.

4. Coast down head and flow versus time will be checked.

5. Level control of pump internal fluid will be checked by varying the cover gas supply at the pump and monitoring pump fluid level.

6. Pump Auxiliary performance (Shaft Seal Lubrication) will be evaluated by measuring seal leak rates during the sustained endurance run.

7. Pump hydraulic performance (head-flow) will be monitored for stability, and vibration levels of the pump will be monitored during the steady state endurance runs.

8. Pump vibration will be monitored during startup and coast down tests.

In addition to the water tests of the prototype pump, a scale model pump is being tested.

B. <u>Prototype Pump Sodium Tests at the Sodium Pump Test Facility</u>

The overall objective of Prototype Pump Testing is to prove capability of the pump to deliver 1000°F sodium at the head and flow conditions specified for the test, and to verify that fluid borne temperature transients up to the limit of the test facility do not cause malfunction (bearing seizure).

Specific objectives are:

1. Demonstrate that the pump is mechanically & hydraulically stable when operated through its full design speed and flow range and to verify hydrostatic bearing performance in the sodium environment.

2. Determine pump hydraulic characteristics (head-flow map and efficiency) in sodium.

3. Demonstrate that high-temperature, and the associated structural temperature gradients do not degrade mechanical operation or hydraulic performance.

4. Demonstrate that the pump and pump auxiliaries are capable of sustained operation while pumping liquid sodium at variable flows and speeds.

5. Demonstrate pump pony motor operation; verify hydrostatic bearing performance in sodium at pony motor speed, demonstrate pony motor developed head at near shut-off, measure head-flow characteristics at different pony motor speeds and different hydraulic loop impedances.

6. Determine any deleterious structural distortion caused by convection in the gas spaces.

7. Demonstrate ability of the pump to withstand sodium fluid temperature transients which simulate predicted plant operating and upset transients.

8. Demonstrate capability of the standpipe-bubbler to maintain adequate sodium level in the pump during steady-state and operating (speed and flow) transients.

9. Verify the pump drive response characteristics with the pump operating in sodium with loop impedance simulating the plant.

10. Demonstrate flow coastdown characteristics (head, flow, speed) from maximum facility flow and from pony motor speed and correlate to similar measurements made in water tests. Determine pump and motor compliance with rotating kinetic energy requirements per E-Specification 22A3444, Table 3.3.1.

11. Measure compliance with Net Positive Suction Head (NPSH) requirements.

61 12. Verify Instrument, Operation, and Maintenance (IOM) Manual procedures for checkout of assembly, operation, disassembly, maintenance, and inspection of pump and auxiliaries.

13. Demonstrate the CRBRP prototype flow controller operatoin with the Drive System and the CRBRP Permanent Magnet (PM) flowmeter.

14. Verify that established rate of dry pump preheat is satisfactory (as indicated by tank and internal temperature gradients).

15. Determine hydraulic impedance of the pump to low magnitude forward flow of sodium through the pump rotor.

16. Confirm performance of the Shaft Seal regarding leak rates.

17. Verify suitability of the pump for subsequent use in sodium after Component Handling and Cleaning Facility (CHCF) cleaning operations.

18. Evaluate whether sodium migration upward or oil migration downward is a concern with the purge of gas feed, labyrinth, and shaft seal arrangement.

19. Determine whether gas injection at the Intermediate Heat Exchanger (IHX) return nozzle causes adverse effects on pump sodium level stability or if slug pumping occurs at the bubbler; and to measure
61 sodium carryover from the bubbler to the gas system.

C. <u>Primary/Secondary Sodium Pumps</u>

Construction and Preoperational Tests (No Sodium)

1. Pump Cover (Canopy) Seal Leakage Tests

The test will be accomplished by injecting helium into the canopy volume and monitoring the inside of the pump for the evidence of leakage by means of a helium detector. The test will prove the adequacy of the seal between the Upper Inner Structure and the tank, i.e., no cover gas leakage into the facility.

2. Electric Power Phasing Check

This test is a check of power phasing wiring from the MG Set to the drive motor. Since high voltages are involved, it will be accomplished with substitute reduced voltages, and commercial phase meters, and verified later by observation of direction of rotation during startup tests.

3. Pump Motor Runout Measurements

The intent of the test is to verify that the pump and motor rotating elements are aligned. The objective is to verify that installed runout is within specifications and to obtain cold (no sodium) shaft torque measurements as a reference value. Commercial runout measurement instrumentation is used.

4. Verify Operation of the Shaft Seal Lube System

The intent of this test is to check out the automatic activation of oil transfer pumps, verify pressurization of tanks, to verify lube lines are filled and flow is achieved, and to verify the leak tightness of the lubrication system. It is accomplished with the

instrument panel which is a part of the Seal Lube System along with some activation procedures and operator action in draining and filling tanks.

5. Preheat Monitoring

The objective of this activity is to verify that the pump heat-up rate does not exceed the specified rate (to prevent applying thermally induced over-stressing) and that temperature gradients throughout the pump do not exceed manufacturers limits.

Verify shaft free rotation before and after preheat.

6. Level Monitoring During Plant Sodium Fill

The intent of this task is to verify that internal fluid level sensors detect rising sodium and that the pump tank/Upper Inner Structure maintains leak tightness, and that pump gas flow is maintained.

D. Plant Preoperational Tests (Coolant Inplace)

1. Pony Motor Sodium Circulation Run

The objectives of this test are:

a. Verify that the shaft seal lubrication system functions properly in automatic actuation of oil transfer pumps, oil leak rate, seal heat exchanger operation, and seal instrumentation.

b. Verify that sodium level variations are detected by the inductive level probes.

c. Verify that the sodium level control (purge gas and standpipe bubbler) system is performing properly.

d. Verify that pump diagnostic instrumentation through the Reactor Heat Transport Instrumentation System readout equipment is performing properly.

e. Verify that the pump rotating assembly operates satisfactorily with respect to vibration at pony motor speed.

f. Verify pony motor operation (vibration, temperature, etc.).

g. Verify shaft seal oil leakage rates are within specification.

h. Verify pump head and flow performance at pony motor speed.

i. Check pump pony motor performance when electrical supply goes to emergency.

j. Provide sodium circulation for sodium purification or other plant checkout.

These tests will be accomplished with several sources of information from the shaft seal lubrication system instrument panel, from the Reactor Heat Transport Instrumentation System for sodium level instruments, shaft position indicators, and vibration measurements, and from plant loop instrumentation for developed head and flow.

2. Pump Speed Control Run

The intent of this test is to verify pump operations with the main drive motor, to verify pony motor clutch engagement, and to operate the loop at higher than pony motor flows for other loop activities.

The objective is to:

a. Verify head/flow as function of speed. Verify shaft seal and lubrication system performance (automatic lube transfer, lube leakage rate, heat exchanger performance, instrumentation) at all main motor speeds.

b. Verify sodium flow rate changes realized from loop flow command.

c. Verify sodium level behavior with speed changes and pump trips.

d. Verify that pump rotating assembly operates satisfactorily regarding vibration.

e. Verify coastdown (head, flow, time) following a trip.

f. Verify proper engagement of the pony motor clutch.

g. Verify motor and motor auxiliaries are performing properly with respect to temperature and lining lubrication.

h. Verify that shaft seal lubrication at 100% speed is satisfactory as indicated by leak rate and automated lube oil transfer.

3. Multipump Tests (coolant inplace, no fuel, but with a dummy core)

The intent of these tests is to verify that multiple loop operation can be satisfactorily achieved by the control system and pumps as indicated by response to speed command, flow command, and trip. Verify that the individual loops are stable, (monitored flow corresponds to flow command) and that hunting does not exist. Determine coastdown characteristics (Head & Flow versus time) following trip. Verify that pressure pulsations from single and combined pumps do not create any undesirable vibrations in the system.

The objective is to:

a. Test the pumps in all expected modes of plant operation.

b. Verify loop flow stability (hunting).

c. Verify pump coastdown (head, flow time) following a trip.

4. Seal Cartridge Replacement Validation

This test consists of replacing a shaft seal from a shutdown pump which has been in the operating plant sodium loop. The intent is to verify procedures and equipment used for seal replacement, and is a validation run where a plant operating environment will require personnel to adhere to the precautions necessary for operating a hot (temperature) system and replacing a critical seal whose life is as such as to require annual replacement.

The objective is to:

Validate the procedures of replacing a shaft seal with loop sodium in place. Demonstrate:

a. Purge.

b. Sodium level control.

c. Cartridge replacement.

d. Post installation checkout.

e. Restoration of pump to service.

This procedure will be accomplished using special handling tools and by either draining the sodium for the pump or by special loop procedures to prevent sodium level from reaching the vicinity of the shaft seal (pressurization, etc.).

5. Pony Motor Plant Operational Test (Fuel in Place)

The intent of this test is to verify that the pony motor operations yields the proper coolant flow for decay heat removal requirements.

The objective is to:

a. Verify that pony motor pumping flow rate is adequate for reactor decay heat removal.

b. Verify leak tightness in pump pit as indicated by radiation sensors (primary only).

c. Verify that temperature gradients in the tank structure resulting from pony motor ON or OFF (flow transients) or reactor temperature changes do not adversely effect pump performance.

These tests will be accomplished using plant radiation sensors, and the Reactor Heat Transport Instrumentation system and the Plant Data Handling and Display System for diagnostic information, and Plant Control System for control.

6. Main Motor Plant Operational Test (Single & Multi Loop Operation)

The intent of this test is to verify that the pump will perform satisfactorily for the plant over the full range of head-flow conditions.

The objectives are to:

a. Verify pump performance at the several plant sodium flow/temperature, conditions.

b. Verify leak tightness in pump pit as indicated by radiation sensors.

c. Provide sodium flow in support of other systems.

14.1.4.3 SECONDARY CONTROL ROD DRIVE

The Secondary Control Rod satisfies the requirement to provide plant protection system shutdown capability which is both redundant and diverse from the Primary Control Rod System.

Prior to installation of plant units there will be extensive tests conducted on components and prototypes. Tests to be conducted on components and prototypes are as follows.

COMPONENT	TEST
Damper	Water Test to 180°F
Latch	Sodium test to 1000°F, to 1125 cycles on 1 unit
Coil Cord	Air Test 1000 cycles of 6 units
Position Indication	Accuracy tests plus long term stability
Control Rod Flow Test	Test in water to 180°F to determine flow splits
Latch Seal	Test in water to 180°F to determine pressure drops
Nosepiece Flow	Test in water to 180°F to determine pressure drops
Prototype Tests	Full scale prototype tests in liquid sodium 400°F to 1000°F to check scram time measurements
Argon Control System	Cycle over the range of temperature expected in plant
	RELIABILITY TESTS
Latch Scram	Accumulation of scram cycles in 1000°F sodium
Latch Real Time	Accumulate long term holds (1 yr.) in 1000°F sodium
System Tests	Test full scale prototypes in liquid sodium 400°F to 1000°F accumulate scram to measure scram times after short and long terms holds

Pneumatic Valve/Cyl.	Test the scram cylinder under temperature and pressures expected in plant
Bellows	Test the driveline bellows under sodium vapor environments at temperatures (400°F - 600°F) expected in plant service

Testing on the secondary control rod drive units will be performed under plant start-up conditions to assure that the integrity of the plant units have not been violated during shipment, handling and installation; to verify that proper installation has been made; and that performance is not adversely affected by fabrication tolerance build-up or by vessel expansion or other thermal effects.

Initially, the housing-to-vessel-nozzle seal will be checked for tightness and the position indicator system will be tested to assure proper functioning and to verify accuracy of rod position indication. Interface conditions will be measured to ensure that inputs to the secondary control rod drives are of a magnitude required to provide adequate secondary control rod response.

Under plant operating conditions, a series of scram tests will be initiated to verify scram time and repeatability, and successful withdrawal and latching functions. Failures, defined as deviation from specification performance values, or inability to perform the scram or latching functions on command, will require removal of the secondary control rod unit, disassembly, analysis of the failure and defining of corrective action. The test series will be repeated using a modified secondary control rod plant unit.

14.1.4.4 UPPER INTERNALS STRUCTURE AND UPPER INTERNALS STRUCTURE JACKING MECHANISM

Acceptance of the Upper Internals Structure (UIS) and Upper Internals Structure Jacking Mechanism (UISJM) as a first-of-a-kind design feature for CRBRP will be made on the basis of scale model testing, operational testing in the vendor's facility and verification of design characteristics during the CRBRP Construction and Preoperational and Startup Testing.

Scale model water testing will be performed to confirm the thermal/hydraulic and vibration adequacy of the UIS and other outlet plenum structures. Flow distribution, pressure drop, and temperature distribution test data will be compared with thermal and hydraulic design criteria for the outlet plenum. Both steady state and transient tests will be performed. Typical outlet plenum structures will be dynamically tested to verify that there will not be adverse vibration during operation.

Prior to operation during the CRBRP Preoperational and Startup Test Program, the UISJM will be tested, as follows. Operation of the UISJM control system will be verified independent of the jacking mechanism. Feedback signals which in normal operation come form strain gauge load sensors, position sensors and limit switches on jacking mechanisms will be electrically simulated into the control system. The jacking mechanism vendor will verify satisfactory operation of the motor, gear and jack for each jacking mechanism independent of the UISJM control system. Finally, testing will be peformed at the UIS vendor to verify that the upper internal structure in combination with the four jacking mechanisms and the jacking mechanism control system can operate in such a manner as to meet the overall system functional requirements. During the installation at the Site, assembly will be checked and alignment verified as part of the CRBRP Construction Test Program. Overtravel limit switches will be set during installation.

The UISJM sealing arrangement will be tested, as follows. Prior to UISJM fabrication, seal leak testing will be conducted on a prototypic seal which includes the buffer O-ring seals and piston rings. Fabrication seal tests will also be performed by the UISJM fabricator. A final leak check of the seals will be performed as part of the CRBRP Construction Test Program after installation at the site.

The operability of the four mechanical jacking systems will be verified in conjunction with the fuel handling operations which will be performed during the CRBRP Preoperation and Startup and Test Program.

Test Phase 1 - Fuel Handling System Operation Check in Air

Test Phase 2 - Fuel Handling System Operation Check in Sodium and Installation of Two Special Core Asemblies

Test Phase 3 - a. Removal of Special Core Assemblies for Inspection

61 b. Initial Fuel Loading

The vibrational behavior of the UIS will be measured during Phase 2 System Operational Tests with sodium flowing through Core Special Assemblies at pony motor flow to 100% and at temperatures of 400°F to 600°F. Significant flow induced lateral, vertical, and torsional structural vibrations will be measured by accelerometers located on the UIS.

Steady state temperature boundary conditions of the UIS will be measured using design verification thermocouples during Phase 4 testing. The temperature data obtained will be used to verify predicted temperature values used in the structural analysis of the UIS.

14.1.4.5 CRBRP INTERMEDIATE HEAT EXCHANGER

A. KEY FEATURE TESTING

The design for the Intermediate Heat Exchanger (IHX) is based upon vendor tests confirmatory of design analyses. These vendor tests were key feature tests. Four specific tests were performed: (1) 30 Degree

Model, (2) 360 Degree Model, (3) Intermediate Flow Test, and (4) Bellows Test.

1) The 30 Degree Model test was peformed to evaluate the shell side flow characteristics and the possibility of tube vibration over the range of IHX operation. This test was an isothermal test on a 30 degree, 20 ft. full scale segment of the tube bundle involving no heat transfer and using water as the testing medium. The results of the test showed the flow paths of the shell side fluid were as predicted analytically and that there was no tube vibration, therefore, confirming analysis of the tube bundle.

2) The 360 Degree Model test was performed to assure that there was uniform flow distribution in the inlet region to the tube bundle. This test was on a 0.3 scale model of the primary inlet plenum of the IHX, including the inlet piping configuration, using water as the testing medium. The results of this test provided the data for designing the inlet baffling to insure balanced flow distribution to the bundle region from 7-1/2% to 100% flow.

3) The Intermediate Flow Test was performed to evaluate the flow distribution at the intermediate inlet to the tube bundle. This test, using a 0.304 scale model and water as the testing medium, was designed specifically to evaluate the intermediate lower (inlet) plenum flow distribution. The results of this test provided the data for designing the baffling to insure balanced flow distribution to the tubes at all flow rates.

4) The Bellows Test was performed to assure that the bellows on the downcomer would survive the design load cycling without failure. This test was a fatigue test which proved the adequacy of the bellows to at least 4 plant lifetimes without failure with a predicted capability to withstand approximately 100 plant lifetimes. A squirm test was also performed which assured the structural adequacy of the expansion members with respect to stability to approximately 2.7 times design pressure.

The results of the above testing confirms the IHX design adequacy.

B. PREOPERATIONAL & STARTUP TESTING

As an integral part of other plant testing and plant power operation, data will be gathered to verify the thermal performance and intermediate side pressure drop characteristics of the IHX.

The thermal performance will be evaluated by measuring the power transferred (Q) and the log mean temperature difference (LMTD) and comparing that with the predicted heat transfer coefficient (UA) using UA = Q/LMTD. The intermediate side pressure drop characteristics can be the intermediate pump discharge and at the intermediate IHX outlet. This measurement, when calculated piping pressure losses between the pressure taps are deleted, will be evaluated against vendor estimates of the intermediate side pressure drop.

The IHX leak tightness and isolation of the primary system from the intermediate system will be demonstrated during the evacuation prior to initial sodium fill and in the sodium inventory observations during Phase 2 testing.

14.1.4.6 STEAM GENERATOR MODULE

A. FEATURE TESTING

The design of the Steam Generator will be supported by several test programs designed to verify assumptions and provide quantitative data to confirm the adequacy of design analyses.

These tests include (1) the Hydraulic Test Model (HTM), (2) Large Leak Tests (LLT), (3) Few Tube Tests (FTT), (4) DNB tests (departure from nucleate boiling), (5) tube support wear tests, (6) material mechanical properties tests. (7) Modular Steam Generator Tests, (8) Single Tube Performance, Stability and Interaction Tests, (9) Tube to Tubesheet Weld Tests, (10) Scale Hydraulic Model Feature Tests, (11) Prototype Steam Generator Tests, and (12) Flow Induced Vibration Tests. See PSAR Section 5.5.3.1.5 for a description of these tests.

B. PREOPERATIONAL AND STARTUP TESTING

A series of tests will be performed on the steam generator modules after they are installed at the site. These tests will be designed to show that the steam generators are properly installed, that they meet all the requirements for safe operation, and that they meet the expected performance requirements.

1. Pre-Operational Tests

   The position and alignment of each module will be checked after it is installed. The module will be checked for leak tightness on both the tube side and the shell side before the sodium and water systems are filled. The water side will be filled first and pressure tested in conjunction with the entire loop (the shell side of the steam generator module will be pressure tested prior to installation). System tests of the water side will provide data on pressure loss vs. flow rate through the module at temperatures up to 400°F. Operability of the module isolation valves and water dump and blowdown subsystem will be tested before the sodium side is filled.

   After all of the IHTS and SGS components are heated to 400°F, the sodium side of the steam generator modules will be filled. System testing of the IHTS will provide data on pressure loss vs. flow rate through the shell side of the steam generator modules.

2. Startup Tests

With the reactor operating, heat transfer and hydraulic performance data will be obtained at several power levels from zero power to 100% of rated power. These data will be used to verify the heat transfer capability and pressure loss calculations. System stability under transient conditions will be used to verify the heat transfer capability and pressure loss calculations. System stability under transient conditions will be demonstrated by changing power levels at the maximum planned rate.

Data will be acquired through Flow Induced Vibration (FIV) instrumentation externally mounted on a superheater and thermal performance in instrumentation built into an evaporator (see section 5.5.3.1.5.1 (K) and (M)).

The objectives of these tests are to:

a) Demonstrate steam generator performance

b) Determine the overall heat transfer coefficient and module pressure losses at rated power and operating conditions.

c) Demonstrate stable operation at low power levels.

d) Demonstrate stable operation at the maximum planned rate of change in power level.

e) Demonstrate the absence of damaging flow induced vibrations.

SCHEDULE FOR INITIAL TEST AND OPERATION

(MONTHS ± CRITICALITY)

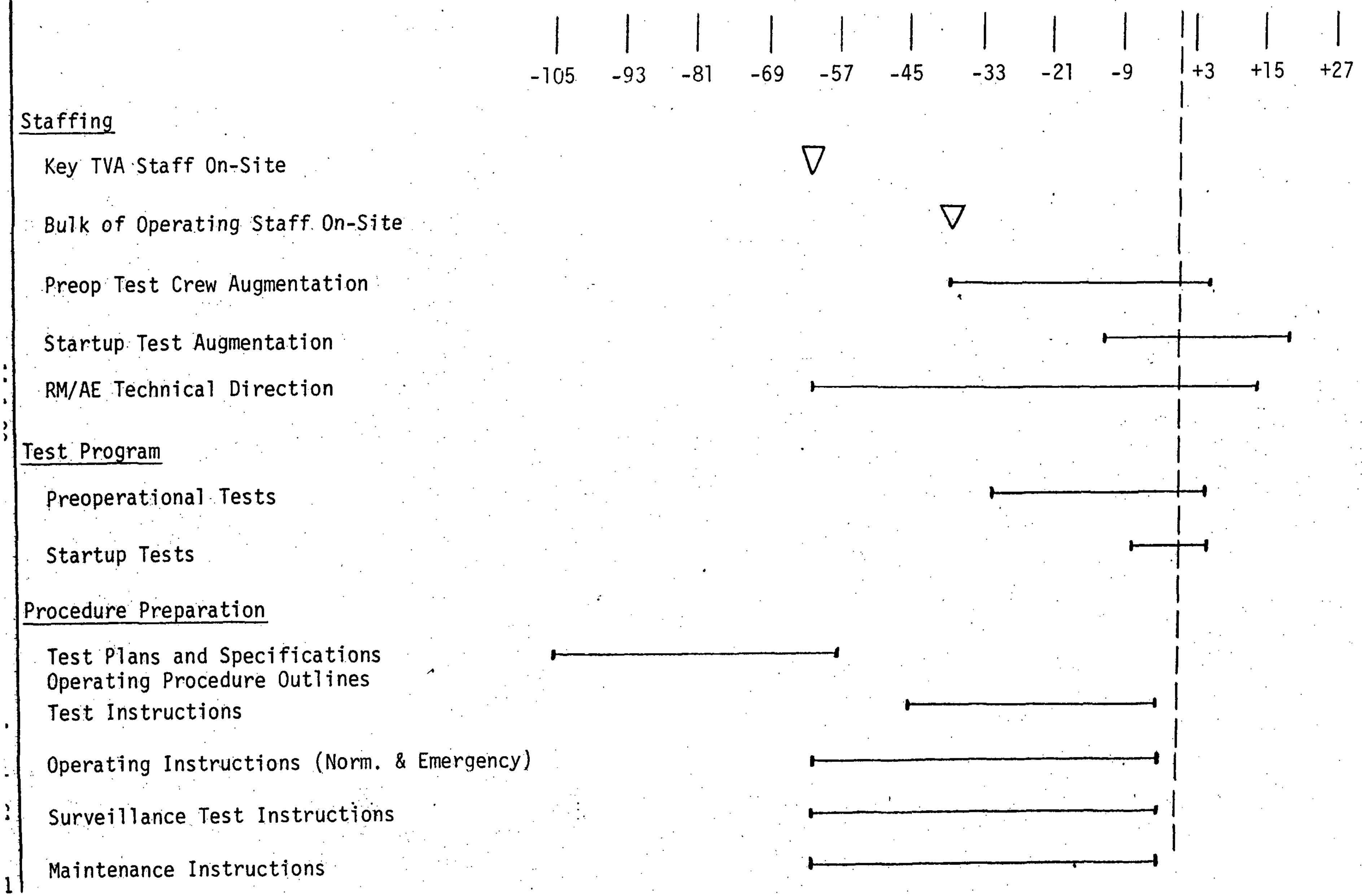

Figure 14.1-1

SCHEDULE FOR INITIAL TEST AND OPERATION

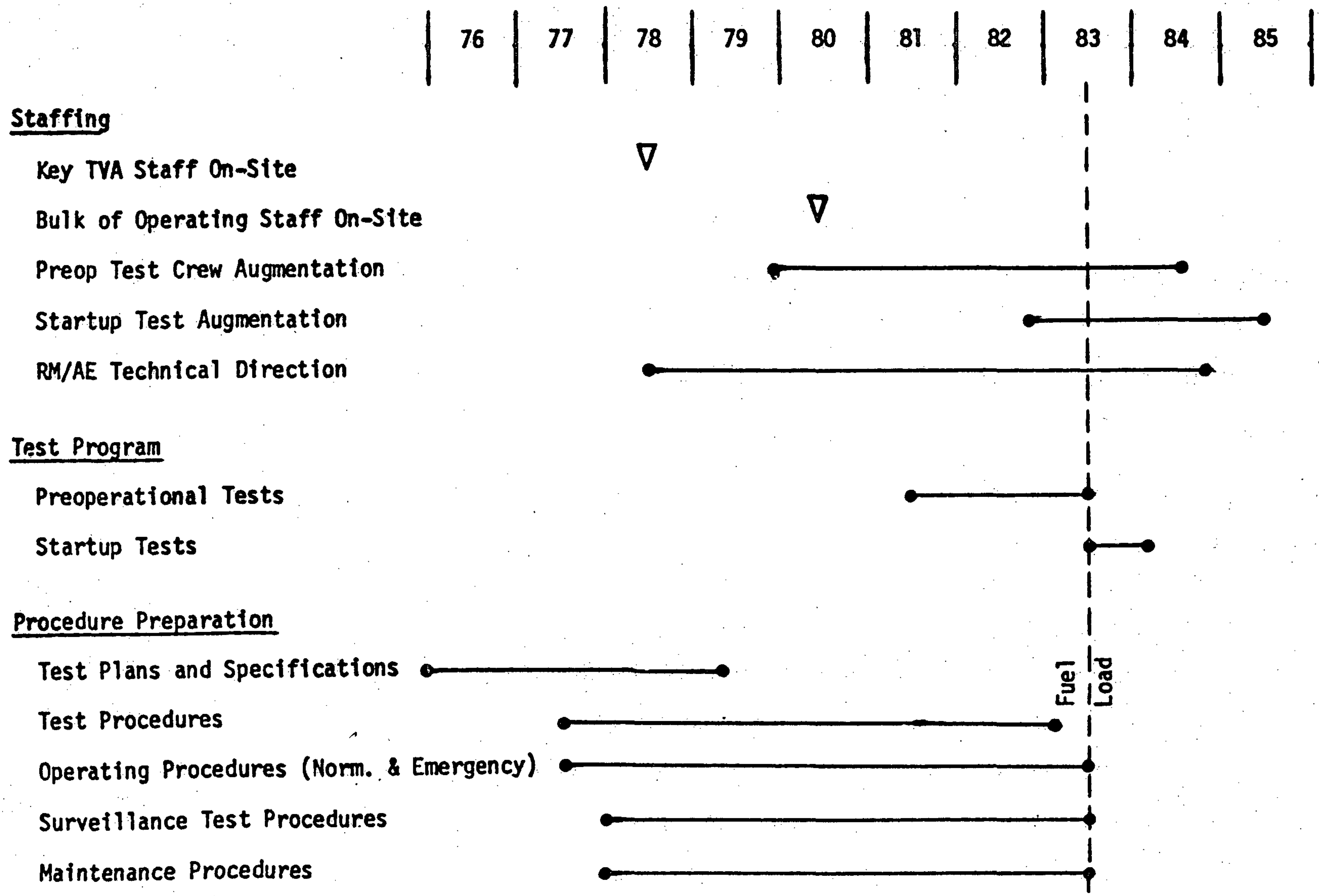

Figure 14.1-1

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14.2 AUGMENTATION OF OPERATOR'S STAFF FOR INITIAL TESTS AND OPERATION

TVA's normal plant operating staff, as described in Section 13.1.2, will be augmented during the initial test and startup period. This augmentation will provide the operating staff with sufficient manpower to safely and effectively conduct the test program, as well as perform those operations functions required during plant startup.

The schedule for providing these additional personnel for augmentation is shown in Figure 14.1-1. Regulatory Guide 1.58 and ANSI Standard N45-2.6-1978 will be used as a guide in developing qualifications of augmenting personnel. The nucleus of the operational testing staff will have had previous experience in testing TVA's light water nuclear plants.

The augmenting personnel described below are in addition to those provided by
61 W-LRM and the A-E for technical direction as described previously in 14.1.3.1.

During the preoperational testing phase the TVA plant operating staff will be augmented with a Preoperational Test Section. This section is an on-site group of TVA employees of the Division of Nuclear Power with the responsibility for the preoperational test program consisting of reviewing test specifications, writing test instructions, assuring that prerequisites
61 are satisfied, conducting the tests, evaluating the test results, and maintaining necessary records of those tests which demonstrate the functional performance and readiness of the various systems. This section is under the direct supervision of the Preoperational Test Program Coordinator who reports to the Plant Manager on functional activities, and the Assistant Director of Nuclear Power (Operations), of the Division of Nuclear Power, on
61 administrative activities.

During the startup testing phase the plant technical staff will be augmented
61 by technical personnel from TVA's Division of Nuclear Power. This includes technical support in nuclear, mechanical, chemical, instrumentation, computer, and general engineering. These technical support personnel will be under the functional supervision of the plant management, and administrative supervision
61 of TVA's Division of Nuclear Power central office.

CLINCH RIVER

BREEDER REACTOR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 15

ACCIDENT ANALYSIS

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15.0 ACCIDENT ANALYSES

15.1 Introduction

15.1.1 Design Approach to Safety

The safety approach for the CRBRP is developed upon three levels of design, a defense-in-depth approach which places the highest importance on achieving safety for the public and plant staff. The three levels of design, the basic differences from the approach used with light water reactor plants, and supporting reliability engineering are described in the section.

The three levels of design emphasize, respectively, quality of design, protection against the consequence of malfunctions, and design features to protect against extremely unlikely faults. In addition, this design incorporates additional features giving assurance of public protection even in the event of an accident beyond the design basis.

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The CRBRP design safety approach is consistent with the three levels
16 of safety concept used by the USNRC to evaluate the adequacy, for licensing purposes, of nuclear power reactors.

The first level focuses on the reliability of operation and prevention of accidents through the intrinsic features of the design, construction, and operation of the plant, including quality assurance, redundancy, testability, maintainability, and failsafe features of the components and systems of the entire plant.

The second level focuses on the protection against "Anticipated Faults" and "Unlikely Faults" which might occur despite the care taken in design, construction, and operation of the plant set forth in level one above. This protection will ensure that the plant is placed in a safe condition following one of these faults.

The third level focuses primarily on the determination of events to be classified as "Extremely Unlikely Faults" and their inclusion in the design basis. These faults are of low probability and no such events are expected to occur during the plant lifetime. Even though they represent extremely unlikely cases of failures, they will be analyzed to establish conservative design bases. In addition to these three levels of design, the CRBRP has included structural and thermal margins for accidents which are beyond the design base (see Section 15.1).

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15.1.1.1 First Level of Design - Inherent and Basic Design Characteristics

An important safety consideration in any reactor is the ability to remove heat from the fuel sufficiently rapidly that the fuel elements do not overheat under any operating or accident conditions. Sodium is an excellent coolant because its favorable combination of viscosity, conductivity, vapor pressure and specific heat provide an excellent intrinsic capability to remove heat. In addition, a sodium-cooled reactor such as the CRBRP operates hundreds of degrees below the boiling point of the coolant. Therefore, the reactor coolants need not be pressurized, the sodium surface above the reactor is at essentially ambient pressure and the pressure exerted on the coolant system boundaries of the plant is only that of the pump head required to force coolant through the reactor. For these reasons, the sodium-cooled reactor has very little stored energy in the coolant; an outstanding advantage compared with systems which operate above the ambient vapor pressure of the fluid at operating temperature, for maintaining system integrity. Small leaks should they occur, have little likelihood of propagation into larger ones. Moreover, the low stored energy in the primary heat transport system does not of itself generate pressure within the secondary containment structure in case of leakage, greatly reducing containment structural requirements relative to those required for light water reactor plants.

In addition to the safety advantages inherent in the use of sodium as the coolant, a number of conceptual and preliminary plant design decisions were made to incorporate design features which avoid the occurrence of accidents or mitigate accident effects should they occur. Examples of these features are:

- A device in each control rod drive mechanism to prevent any rapid outward motion of rods.
- Provisions to prevent gas from entering the reactor core, including:
  - A vortex suppressor to prevent gas entrainment at the reactor vessel free surface, and
  - Continuous bleeding of small bubbles from the system.
- A thermal liner in the reactor vessel to maintain the upper vessel walls 100 to 150°F cooler than the reactor outlet temperature and protect them from thermal transients associated with power level changes.
- Selection of core materials to give a negative Doppler coefficient of reactivity and thus provide a reliable feedback mechanism enhancing stability in normal operation and limiting reactivity excursions.
- Reactor fuel subassemblies with fuel pin spacing designed to reduce potential for reductions in coolant flow due to fuel swelling.

- Coordinated mechanical design of core assembly, core support, and fuel handling machine control system to assure that a subassembly cannot be positioned by the fuel handling machine in a location of increased reactivity or of reduced flow.

- Core support structure inlet modules and assembly inlet nozzles which provide multiple inlet passages and also prevent passage of foreign material greater than 1/4 inch across to prevent flow blockage.

The project is to use, to the maximum extent practicable, proven technology, including the incorporation of applicable FFTF, light water reactor, and other nuclear power experience. Where this technology and experience are not applicable or are only partially or indirectly applicable, an extensive program of development and proof tests is being implemented.

15.1.1.2 Second Level of Design - Protection Against Anticipated and Unlikely Faults

Recognizing that errors, or malfunctions can occur despite the care and attention given to the plant design, construction, operation and maintenance, two avenues of second level pursuit have been followed: (1) a number of protective systems and plant features have been provided to protect against malfunctions, and to limit their consequences to definable and acceptable levels, and (2) a program of development and testing has been undertaken to define clearly the nature and consequences of accidents such as fuel failure, which might result from malfunctions. These features are:

- The plant protection system provides prompt automatic shutdown of the reactor when necessary to correct for off-normal conditions in the system. Two redundant, independent fast-acting systems are provided. Each system is complete with diverse sensors, logic, and circuitry, and each actuates separate, diverse sets of neutron absorber rods.

- All systems, components, and structures required for continued safe operation are designed to withstand or be protected from the effects of abnormal environmental conditions, such as earthquakes or floods.

- The three-loop design provides a redundant heat removal system such that core cooling is maintained even if, at the same time as a loss of normal power, an active component of one loop is disabled.

- Pony motors are provided for the primary and intermediate loop pumps of the heat transport system. They engage automatically upon reactor scram or shutdown to provide forced coolant circulation. The pony motors are capable of receiving power from the standby diesel generators.

- Natural circulation capability is provided in both primary and intermediate loops of the Heat Transport System.

- Extensive sodium leak detection capability is provided to assure that any failure of the primary boundary is detected promptly so that corrective action can be taken.

- A Heat Removal System of completely independent flowpath exists which uses the makeup and overflow system of the reactor vessel and rejects heat to the ex-vessel fuel storage system.

- The primary system components of each of the three independent heat transfer systems is installed in an isolable massive reinforced concrete, steel-lined, inerted cell.

- A sensitive and redundant system to detect the initiation of small leaks in the steam generator modules.

- A steam generator pressure relief system which handles reaction products in the event of a large leak.

- Guard vessels and elevated piping assure core coverage and continuity of core cooling even in the event of primary coolant system leaks.

The design emphasizes in this second level the need to insure and confirm the high reliability of these protection systems and of any component or system whose failure could lead to severe core damage. An extensive program of qualitative and quantitative analysis and development testing is underway to enable the Project to base its design of public public protection on the surety of these protection systems.

The basic objective of the Reliability Program is to provide additional assurance (beyond the normal design process) that the RSS and SHRS can be expected to perform their intended functions. Based on a review of licensing requirements and associated Regulatory Guides currently in use on thermal reactors and an evaluation of potential sources of release of radiological species, it was considered prudent to devote the additional effort to safety related systems which provide prevention of loss of coolable geometry in the reactor core. The focus of reliability activities is, therefore, placed on confirmation that the reactor shutdown systems and shutdown heat removal system are highly reliable since these systems are most important to prevention of loss of coolable core geometry.

The Reliability Program, as described in Appendix C, emphasizes 1) reliability enhancement through qualitative and quantitative analyses of components and systems which comprise and interface with the shutdown and shutdown heat removal systems and 2) reliability verification through component and system level tests under both design and overload conditions. Reliability activities include reliability requirements placed in appropriate design documents, Reliability Engineering review and approval of design documents and generation of Reliability Design Support Documents to contain reliability analyses and results of testing evaluations. These activities are directed towards providing feedback into the design of shutdown and shutdown heat removal system components to assure their reliable 3
41 operation.

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15.1.1.3 Third Level of Design

The third level of design provides an extra measure of protection for the public health and safety, beyond that provided by the first and second levels, by imposing design requirements derived from low probability events. This is done in two stages:

Extremely Unlikely Faults are included as design basis events. The plant design must include appropriate safeguard features to accommodate all of these events. Typical conservative assumptions, such as failure of a single component, are used in the analysis of these faults to demonstrate adequate design protection. Analytic evaluations of the capability of the plant to withstand the identified Extremely Unlikely Faults are presented in this Chapter.

15.1.1.4 Margins Beyond the Design Base

In addition to the three levels of design discussed above a further extra measure of protection for the public health and safety has been provided by imposing structural and thermal margin requirements on the plant design which are derived from a spectrum of events which lie beyond the plant design base. The Structural Margins Beyond the Design Base (SMBDB) impose additional structural loadings (based on HCDA analyses) on the reactor vessel system and PHTS components and assure that extra margins exist to acceptably accommodate the additional requirements over and above those of the design basis accidents. The Thermal Margins Beyond the Design Base (TMBDB) address the meltdown sequences that could follow an HCDA and assure that the radiological consequences would be accommodated and/or mitigated to acceptable levels. Details and evaluations of the plant capabilities in these regards are provided in References 10a and 10b, PSAR Section 1.6.

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15.1.2 Requirements and Criteria for Assessment of Fuel and Blanket Rod Transient Performance

To assure that the CRBRP fuel and blanket rods will operate safely over their respective design lives, the qualitative requirements of Table 15.1.2-1 have been implemented. Specifically, as discussed in Chapter 4, mechanical design limits (cumulative damage function and ductility limited strain) have been developed to assure cladding integrity is maintained through normal operation, all Anticipated Faults and the most severe Unlikely Fault. The complete details of cumulative mechanical damage function including both the theoretical derivation and experimental data base, are provided in Reference 58 of Section 4.2. For Extremely Unlikely Faults, limits have been established in terms of cladding and coolant temperatures to conservatively ensure that core coolable geometry will be maintained. All these limits taken together ensure that the requirements of Table 15.1.2-1 are met.

The acceptance criteria of Table 15.1.2-2 have been developed for the preliminary safety review (PSAR) to evaluate the acceptability of each transient analyzed relative to the requirements of Table 15.1.2-1. The use of the acceptance criteria allows preliminary assessment of each transient event without the need for detailed calculations of mechanical damage. Detailed calculations of mechanical damage will be performed for the final safety review (FSAR). The following subsections provide a brief description of the acceptance criteria of Table 15.1.2-2.

15.1.2.1 Acceptance Criteria for Anticipated and Unlikely Faults

The preliminary acceptance criteria for the Anticipated and Unlikely faults were established to assure that cladding integrity is maintained throughout all Anticipated Faults in the fuel (or blanket) lifetime and limiting-case Unlikely Fault. As indicated in Table 15.1.2-2, faults are considered acceptable if there is no fuel melting and the maximum cladding temperatures are less than 1500°F for Anticipated Faults and 1600°F for Unlikely Faults. These criteria not only assure that the lifetime cladding performance requirements are met, they also assure a large margin to sodium boiling and to cladding melting, thereby assuring that core coolable geometry will be maintained.

The bases for the preliminary acceptance criteria are the analyses of mechanical performance of fuel and blanket rods discussed in Chapter 4. Specifically, worst case umbrella transients corresponding to the thermal conditions in the criteria have been combined at the required frequency at the worst time in life. In each case the mechanical design limits, cumulative mechanical damage function and ductility limited strain (see Section 4.2.1) have been satisfied. Therefore, if the Anticipated or Unlikely Fault produces lower fuel and cladding temperatures than those used in the corresponding umbrella transients (and preliminary acceptance criteria), then the design performance
61 and safety objectives of the fuel and blanket rods are satisfied.

The actual mechanical damage to the cladding is a complicated function of temperature, stress (fuel and fission gas expansion), time at that stress, and accumulated irradiation (reduced strength and ductility). Therefore, violation of the preliminary acceptance criteria on fuel temperature, cladding temperature or both does not necessarily mean the transient has unacceptable damage. It does require the calculation of mechanical damage in accordance with the design procedures described in Chapter 4. The fault is unacceptable only if the corresponding damage violates the mechanical design limits given in Section 4.1, or if the criteria for Extremely Unlikely Faults are not met. Should events occur which are more severe than the transient events used as the design envelope, then the records of actual core environmental conditions will be utilized to determine the actual cumulative damage function; which would then be compared to the design limits.

15.1.2.2 Acceptance Criteria for Extremely Unlikely Faults or Postulated Accidents

The allowable limit for an Extremely Unlikely Fault is defined as maintaining coolable geometry. As indicated in Table 15.1.2-2 events of this type are considered acceptable if the coolant temperature remains below boiling and the cladding temperature remains below melting.

The basis for the acceptance criteria on Table 15.1.2-2 for these types of faults is that the geometry of the core must remain coolable following a faulted event to assure that damage will not progress. This limit is considered to be met when the cladding temperature is held below the melting point. If there is no cladding melting then no gross cladding relocation or gross channel blockage can occur. Therefore, preventing cladding temperatures from exceeding the melting temperature will ensure maintaining a coolable core geometry.

Before the cladding melting temperature can be reached, it is necessary to first experience bulk sodium boiling and then dryout of the cladding. The prevention of sodium boiling is considered as a necessary and sufficient criterion for ensuring a core coolable geometry.

15.1.2.3 Acceptance Criteria Dependence on Shutdown Mode

As noted in Table 4.2-35 in Chapter 4, the next higher level of damage is allowed for secondary shutdown system event termination. The rationale is that failure to actuate the primary shutdown system is a low probability event so that the combined probability of the event occurring and secondary shutdown system activation being required is much lower than the probability of the event occurring. Therefore, application of the acceptance criteria of Table 15.1.2-2 in the safety analyses reported in this chapter considers shutdown by the Primary and Secondary Shutdown System
61 action separately in a manner as described in Table 4.2-35.

TABLE 15.1.2-1

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EVENT CLASSIFICATION AND DAMAGE SEVERITY LIMITS

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Event Classification: Mechanical Design (Chapter 4)	Event Classification: RDT Standard C-16-1	Severity Level: RDT Standard C-16-1
Normal: Any condition of system startup, design range operations, hot standby, or shutdown other than an upset, emergency, faulted or testing conditions.	Normal Operation: Normal operation includes steady power operations and those departures from steady operation which are expected frequently or regularly in the course of power operations, refueling, maintenance, or maneuvering of the plant.	No Damage: No damage is defined as 1) no significant loss of effective fuel lifetime; 2) accommodations within the fuel and plant operating margins without requiring automatic or manual protective action; and 3) no planned release of radioactivity.
Upset: Any abnormal incident not causing a forced outage or causing a forced outage for which the corrective action does not include any repair of mechanical damage.	Anticipated Faulted: An off-normal condition which individually may be expected to occur once or more during the plant lifetime.	Operational Incident: An operational incident is defined as an occurrence which results in 1) no reduction of effective fuel lifetime below the design values; 2) accommodation with, at most, a reactor trip that assures the plant will be capable of returning to operation after corrective action to clear the trip cause; and/or 3) plant radioactivity releases that may approach the 10CFR20 guidelines.
Emergency: Infrequent incident requiring shutdown for correction of the condition or repair of damage in the system. No loss of structural integrity.	Unlikely Faulted: An off-normal condition which individually is not expected to occur during the plant lifetime; however, when integrated over all plant components, events in this category may be expected to occur a number of times.	Minor Incident: A minor incident is defined as an occurrence which results in 1) a general reduction in the fuel burnup capability and, at most, a small fraction of fuel rod cladding failures; 2) sufficient plant or fuel rod damage that could preclude resumption of operation for a considerable time and/or 3) plant radioactivity releases that may exceed 10CFR20 guidelines, but does not result in interruption or restriction of public use of areas beyond the exclusion boundary.
Faulted: Postulated event and consequences where integrity and operability may be impaired to the extent that considerations of public health and safety are involved.	Extremely Unlikely Faulted: An off-normal condition of such extremely low probability that no events in this category are expected to occur during the plant lifetime, but which nevertheless represents extreme or limiting cases of failures which are identified as design bases.	Major Incident: A major incident is defined as an occurrence which results in 1) substantial fuel and/or cladding melting or distortion in individual fuel rods, but the configuration remains coolable; 2) plant damage that may preclude resumption of plant operations, but no loss of safety functions necessary to cope with the occurrence; and/or 3) radioactivity release that may exceed the 10CFR20 guidelines but are well within the 10CFR100 guidelines.

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TABLE 15.1.2-2

ACCEPTANCE CRITERIA FOR PRELIMINARY SAFETY EVALUATION

Event Classification	Severity Level[4]	Fuel Temperature	Cladding Temperature (°F)	Coolant Temperature (°F)
Anticipated Fault	Operational Incident	Solidus[1],[2]	1500[1]	N/A
Unlikely Fault	Minor Incident	Solidus[1],[2]	1600[1]	N/A
Extremely Unlikely Fault or Postulated Accident	Major Incident	---	Solidus (2475)	Saturation[3]

NOTES:

(1) For temperatures in excess of these values, transients shall be assessed using mechanical design procedures and design limits of Chapter 4.2.

(2) No fuel melting at existing conditions.

(3) No sodium boiling at existing pressure.

(4) Applicable "Event Class" or "Severity Level" is based on Primary Shutdown System
61 action . For Secondary System Shutdown see Table 4.2-35.

in which the system is shifted into the plastic region an additional CDF component is required. This component accounts for the hysteresis effect typically encountered when creep rupture formulations are applied (extrapolated) in the vicinity of the ultimate strength.

In terms of the plastic-transient component the entire CDF is given by

$$\Delta L(t) = \int_0^t dt/t_r[T(t),\sigma(t),C(t),F(t)]+L_T[\sigma(t),\sigma_u(T,C,F),\sigma_p(T,C,F)], \quad (8)$$

where L_T is an empirical function describing the CDF (Life Fraction) worth of a stress into the plastic region. This function, based on HEDL data with prototypic FFTF clad tubing (Ref. 3), is shown in Figure 15.1.2-3.

In Equation (8) σ_u and σ_p are functions describing the ultimate strength and proportional elastic limit, respectively. These are dependent on temperature, composition and fluence and, therefore, L_T is time dependent according to

$$\dot{L}_T(t)=(\delta L_T/\delta\sigma)\dot{\sigma}+(\delta L_T/\delta\sigma_u)\{(\delta\sigma_u/\delta T)\dot{T}+(\delta\sigma_u/\delta C)\dot{C}+(\delta\sigma_m/\delta F)\dot{F}\}$$

$$+(\delta L_T/\delta\sigma_p)\{(\delta\sigma_p/\delta T)T+(\delta\sigma_p/\delta C)C+(\delta\sigma_p/\delta F)F\} \quad (9)$$

The functions describing the temperature dependence of the proportional elastic limit and the ultimate strength of irradiated, prototypic tubing are shown in Figures 15.1.2-4 and -5, respectively.

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15.1.3 Control Rod Shutdown Rate and Plant Protection System Trip Settings

15.1.3.1 Control Rod Shutdown Rate

The specifications and assumptions that define the time at which scram is initiated, together with the rate of negative reactivity insertion during scram, are significant factors in many of the analyses presented in this Chapter. The following paragraphs address the control rod shutdown rate considerations.

The selection of the numbers and locations of the control rods assigned to the primary and secondary shutdown systems was based on analyses of "worst case" minimum shutdown margin and minimum rod worth analyses (see Section 4.3.2). The reactivity insertion rates of the selected banks of rods are then calculated on the basis of assumed rates of rod motion, with the highest worth rod in the bank assumed stuck at the pre-scram position (see Section 4.2.3).

Primary Shutdown System

A minimum primary system scram insertion rate requirement was developed to assure adequate shutdown capability against design basis transients in the event of maximum time delays, minimum rod insertion velocities, and minimum control rod reactivity worths. Subsequently, the expected performance of the primary mechanical shutdown system was established based on extensive testing of prototype systems. Comparisons of minimum performance requirements with the expected performance are given in Section 4.2.3.

The expected scram rate utilizes predicted primary control rod start of cycle positions and nominal rod worths based on values normalized to the results of CRBRP related critical experiments. By applying start of cycle shutdown rates, a conservative insertion rate is utilized since the primary control rod start of cycle positions and nominal rod worths are based on values normalized to the results of CRBRP related critical experiments over the operating cycle as rods are withdrawn to compensate for fuel burnup. Section 4.2.3 describes the variations in scram insertion rates for the primary system. The beginning of equilibrium cycle insertion rate has been applied in this chapter since the equilibrium cycle fuel conditions tend to be more limiting than the early cycle conditions (e.g., the maximum power fuel assembly exists for this core condition).

Results of analyses for worst case undercooling and overpower transients, utilizing minimum insertion rate requirements, are presented in this chapter and compared to results obtained by using the expected insertion rate. These transients include a loss of off-site electrical power (Section 15.3.1) and a wide parametric range of step reactivity insertions occuring due to the Safe Shutdown Earthquake (Section 15.2.3.3) and loss of hydraulic holddown (Section 15.2.2.1). Other transients in this chapter are not sensitive to the insertion rates and have been evaluated using only the expected insertion rate to provide the most realistic assessment of the events.

Secondary Shutdown System

Since the secondary shutdown system control rods are always fully withdrawn under operating conditions, the scram insertion rates do not vary significantly over the operating cycle. A preliminary minimum insertion rate requirement has been developed for the secondary system and is given in Section 4.2.3. This requirement must be satisfied for minimum rod worths (See Section 4.3.2) and maximum withdrawal limits on control rod positions (see Section 4.2.3). The expected insertion rates exceed this minimum as shown in Section 4.2.3. The minimum insertion rate requirement has been used for all secondary shutdown system results reported in this chapter.

TABLE 15.1.3-1

PPS SUBSYSTEM TRIP LEVELS OR TRIP EQUATIONS

Primary Shutdown System

High Flux	Trip at 115% power
Flux-Rate	Positive: $\mathcal{L}^{-1}\left[\phi(s) * \frac{1.01}{1+28S}\right] - 0.99\phi(t) + 0.1706Np + 0.0364 \le 0$ Negative: $1.01\phi(t) - \mathcal{L}^{-1}\left\{\phi(s)\ I\ \frac{1.01}{1+28s}\ 0.1969Np\right\} + 0.0416 \le 0$
Flux to Pressure	$1.318\sqrt{P} - \phi + 0.0425 \le 0$
Primary to Intermediate Speed Ratio	$Np\,(0.147 \pm 0.0022) + 0.0595 \pm 0.0007 - \text{AbsVal}\,[Np\,(1 \pm 0.015) - N_I\,(1 \pm 0.015) + 0.0075 \pm 0.01] \le 0$
HTS Pump Frequency	Trip at 57 Hertz
Reactor Vessel Level	Trip when level drops 18" from normal operating level
Steam to Feedwater Flow Ratio	Trip at 30% mismatch
IHX Primary Outlet Temperature	Trip at 830°F

Secondary Shutdown System

Flux to Total Flow	$1.2\Sigma Fp - 0.99\,\phi + 0.087 \le 0$
Startup Flux	Trip before 10% power
Primary to Intermediate Flow Ratio	$Fp\,(0.147 \pm 0.0022) + 0.050 \pm 0.0007 - \text{abs Val}\,[Fp\,(1 \pm 0.015) - F_I\,(1 \pm 0.015) + 0.0075 \pm 0.01] \le 0$
Steam Drum Level	Trip at 8" drop from full power steady state level

TABLE 15.1.3-1 (Continued)

High Evaporator Outlet Temperature	Trip at 750°F
Sodium Water Reaction	Trip initiated within 3.0 seconds
HTS Pump Voltage	Trip at 75% of rated voltage

Definition of Variables

$\mathcal{L}^{-1}$ = Laplace Operator

ϕ = Reactor Flux

Np = Average Primary Pump Speed

Np = Primary Pump Speed

N_I = Intermediate Pump Speed

P = Reactor Inlet Plenum Pressure

ΣF_p = Total Primary Pump Flow

F^p = Primary Pump Flow

F_I^p = Intermediate Pump Flow

The above variables are normalized such that their value at 100% conditions = 1.0.

TABLE 15.1.3-2

ACCIDENT EVENTS

PSAR Section	Anticipated Events
15.2.1.1	Control Assembly Withdrawal at Startup
15.2.1.2	Control Assembly Withdrawal at Full Power
15.2.1.3	Seismic Reactivity Insertion (OBE)
15.2.1.4	Small Reactivity Insertions
15.2.1.5	Inadvertent Drop of a Single Control Rod at Full Power
15.3.1.1	Loss of Off-Site Electrical Power
15.3.1.2	Spurious Primary Pump Trip
15.3.1.3	Spurious Intermediate Pump Trip
15.3.1.4	Inadvertent Closure of One Evaporator or Superheater Module Isolation Valve
15.3.1.5	Turbine Trip
15.3.1.6	Loss of Normal Feedwater
15.3.1.7	Inadvertent Actuation of the Sodium/Water Reaction Pressure Relief System
15.7.1.1	Loss of One D.C. System
15.7.1.2	Loss of Instrument or Valve Air System
15.7.1.3	IHX Leak
15.7.1.4	Off-Normal Cover Gas Pressure in PHTS
15.7.1.5	Off-Normal Cover Gas Pressure in IHTS
	Unlikely Events
15.2.2.1	Loss of Hydraulic Holddown
15.2.2.2	Sudden Core Radial Movement
15.2.2.3	Maloperation of Reactor Plant Controller
15.3.2.1	Single Primary Pump Seizure
15.3.2.2	Single Intermediate Loop Pump Seizure
15.3.2.3	Small Water to Sodium Leaks in Steam Generator Tubes
15.3.2.4	Failure of Steam Bypass System
15.5.2.1	Fuel Assembly Dropped During Refueling
15.5.2.2	Attempt to Insert a Fuel Assembly into an Occupied Position
15.5.2.3	Single Fuel Assembly Cladding Failure in Fuel Handling System
15.5.2.4	Cover Gas Release During Refueling
15.5.2.5	Heaviest Crane Load Impacts Reactor Closure Head
15.7.2.1	Inadvertent Release of Oil Through the Pump Seal (PHTS)
15.7.2.2	Inadvertent Release of Oil Through the Pump Seal (IHTS)
15.7.2.3	Generator Breaker Failure to Open at Turbine Trip
15.7.2.4	Rupture in the RAPS Cold Box
15.7.2.5	Liquid Radwaste System Failure
15.7.2.6	Failure in the EVST NaK System
15.7.2.7	Leakage from Sodium Cold Traps
15.7.2.8	Rupture in RAPS Noble Gas Storage Vessel Cell
15.7.2.9	Rupture in the CAPS Cold Box

TABLE 15.1.3-2 (Continued)

Extremely Unlikely Events

15.2.3.1	Cold Sodium Insertion
15.2.3.2	Gas Bubble Passage Through Core, Radial Blanket and Control Assembly
15.2.3.3	Core, Radial Blanket, and Control Rod Movement Due to Safe Shutdown Earthquake
15.2.3.4	Control Assembly Withdrawal at Startup - Maximum Mechanical Speed
15.2.3.5	Control Assembly Withdrawal at Power - Maximum Mechanical Speed
15.3.3.1	Steam or Feed Line Pipe Break
15.3.3.2	Loss of Normal Shutdown Cooling System
15.3.3.3	Large Sodium/Water Reactions
15.3.3.4	Primary Heat Transport System Pipe Leak
15.3.3.5	Intermediate Heat Transport System Pipe Leak
15.5.3.1	Collision of EVTM with Control Rod Drive Mechanisms
15.6.1.1	Primary Sodium In-Containment Storage Tank Failure During Maintenance
15.6.1.2	Failure of the Ex-Vessel Storage Tank Sodium Cooling System During Operation
15.6.1.3	Failure of Ex-Containment Primary Sodium Storage Tank
15.6.1.4	Primary Heat Transport System Piping Leaks
15.6.1.5	Intermediate Heat Transport System Piping Leak
15.7.3.1	Leak in a Core Component Pot
15.7.3.2	Spent Fuel Shipping Cask Drop from Maximum Possible Height
15.7.3.3	Maximum Possible Conventional Fires, Flood or Storms or Minimum River Level
15.7.3.4	Failure of Plug Seals and Annuli
15.7.3.5	Fuel Rod Leakage Combined with IHX and Steam Generator Leakage
15.7.3.6	Sodium Interaction with Chilled Water
15.7.3.7	Sodium-Water Reduction in Large Component Cleaning Vessel

TABLE 15.1.3-3
SYSTEMS ASSUMED OPERABLE TO MITIGATE THE CONSEQUENCES FOLLOWING THE OCCURRENCE
OF EACH ACCIDENT EVENT

Events		Required Operable System	Primary	Secondary
15.2.1	Anticipated Events			
15.2.1.1	Control Assembly Withdrawal at Startup	PPS followed in long term by decay heat removal (1)	Flux-Pressure Flux-Delayed Flux	Flux-Total Flow
15.2.1.2	Control Assembly Withdrawal at Power	PPS followed in long term by decay heat removal	High Flux Flux-Pressure	Flux-Total Flow
15.2.1.3	Seismic Reactivity Insertions-OBE	PPS followed in long term by decay heat removal	High Flux Flux-Pressure	Flux-Total Flow
15.2.1.4	Small Reactivity Insertions	PPS followed in long term by decay heat removal	High Flux	Flux-Total Flow
15.2.1.5	Inadvertent Drop of a Single Control Rod at Full Power	PPS followed in long term by decay heat removal	Flux-Delayed Flux	Modified Nuclear Rate
15.2.2	Unlikely Events			
15.2.2.1	Loss of Hydraulic Holddown	PPS followed in long term by decay heat removal	High Flux Flux-Pressure	Flux-Total Flow
15.2.2.2	Sudden Core Radial Movement	PPS followed in long term by decay heat removal	High Flux Flux-Pressure	Flux-Total Flow
15.2.2.3	Maloperation of Reactor Plant Controllers	PPS followed in long term by decay heat removal	High Flux Flux-Pressure	Flux-Total Flow

TABLE 15.3-3 (Continued)

Events		Required Operable System	Primary	Secondary
15.2.3	Extremely Unlikely Events			
15.2.3.1	Cold Sodium Insertion	PPS followed in long term by decay heat removal	Speed Ratio	Flow Ratio
15.2.3.2	Gas Bubble Passage through Fuel, Radial Blanket and Control Assemblies	PPS followed in long term by decay heat removal	High Flux	Flux-Total Flow
15.2.3.3	Seismic Reactivity Insertion-SSE	PPS followed in long term by decay heat removal	High-Flux Flux- Pressure HTS Pump Electrics	Flux-Total Flow
15.2.3.4	Control Assembly withdrawal at Startup-Maximum Mechanical Speed	PPS followed in Long term by decay heat removal	Flux- Pressure Flux-Delayed Flux	Flux-Total Flow
15.2.3.5	Control Assembly Withdrawal at Power	PPS followed in long term by decay heat removal	High Flux	Flux-Total Flow
15.3.1	Anticipated Events			
15.3.1.1	Loss of Off-Site Electric Power	PPS followed in long term by decay heat removal	HTS Pump Frequency	HTS Pump Voltage
15.3.1.2	Spurious Primary Pump Trip	PPS followed in long term by decay heat removal	Flux to Pressure Speed Ratio	Flow Ratio
15.3.1.3	Spurious Intermediate Pump Trip	PPS followed in long term by decay heat removal	Speed Ratio	Flow Ratio
15.3.1.4	Inadvertent Closure of One Evaporator or Superheater Module Isolation Valve	PPS followed in long term by decay heat removal	Steam-Feedwater	Evap. Outlet Temp.
15.3.1.5	Turbine Trip	Long Term by decay heat removal (2)	Steam-Feedwater	Loss of Condenser Vacuum

TABLE 15.1.3-3 (Continued)

Events		Required Operable System	Primary	Secondary
15.3.1.6	Loss of Normal Feedwater	PPS followed in long term by decay heat removal	Steam-Feedwater	Steam Drum Level
15.3.1.7	Inadvertent Actuation of the Sodium-Water Reaction Pressure Relief System	PPS followed in long term by decay heat removal	Steam-Feedwater	Evap. Outlet Temp.
15.3.2	Unlikely Events			
15.3.2.1	Single Primary Pump Seizure	PPS followed in long term by decay heat removal	Speed Ratio	Flow Ratio
15.3.2.2	Single Intermediate Loop Pump Seizure	PPS followed in long term by decay heat removal	Speed Ratio	Flow Ratio
15.3.2.3	Small Water-to-Sodium Leaks in Steam Generator tubes	(3)		
15.3.2.4	Failure of the Steam Bypass System	PPS followed in long term by decay heat removal	Steam-Feedwater	Steam Drum Leval
15.3.3	Extremely Unlikely Events			
15.3.3.1	Steam or Feed Line Pipe Break	PPS followed in long term by decay heat removal	Steam-Feedwater	Evap. Outlet Temp.
15.3.3.2	Loss of Normal Shutdown Cooling System	PPS followed in long term by decay heat removal	Steam-Feedwater	Steam Drum Level
15.3.3.3	Large Sodium-Water Reaction	Sodium water reaction pressure relief system rupture discs	Steam-Feedwater	Sodium Water Reaction
15.3.3.4	Primary Heat Transport System Pipe Leak	(3)		
15.3.3.5	Intermediate Heat Transport System Pipe Leak	PPS followed in long term by decay heat removal	IHX Pri Outlet	Flow Ratio

Events		Required Operable System	Primary	Secondary
15.5.2	Unlikely Events			
15.5.2.1	Fuel Assembly Dropped within Reactor Vessel during Refueling	(3)		
15.5.2.2	Damage of Fuel Assembly due to Attempt to Insert a Fuel Assembly Into an Occupied Position	(3)		
15.5.2.3	Single Fuel Assembly Cladding Failure and Subsequent Fission Gas Release during Refueling	EVTM Seals		
15.5.2.4	Cover Gas Release during Refueling	(3)		
15.5.2.5	Heaviest Crane Load Impacts Reactor Closure Head	(3)		
15.5.3	Extremely Unlikely Events			
15.5.3.1	Collision of EVTM with Control Rod Drive Mechanism	(3)		
15.6	Sodium Spills			
15.6.1.1	Primary Sodium In-Containment Storage Tank Failure during Maintenance	Containment Isolation System		
15.6.1.2	Failure of the Ex-Vessel Storage Tank Sodium Cooling System during Refueling	(3)		
15.6.1.3	Failure of Ex-Containment Primary Sodium Storage Tank	(3)		

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TABLE 15.3-3 (Continued)

Events		Required Operable System	Primary	Secondary
15.6.1.4	Primary HTS Pipe Leak	(3)		
15.6.1.5	Intermediate HTS Pipe Leak	(3)		
15.7.1.1	Loss of D.C. System	(3)		
15.7.1.2	Loss of Instrument Or Valve Air	(3)		
15.7.1.3	IHX Leak	(3)		
15.7.1.4	Off-Normal Cover Gas Pressure in the Reactor Coolant Boundary	(3)		
15.7.1.5	Off-Normal Cover Gas Pressure in the IHTS	(3)		
15.7.2.1	Inadvertent Release of Oil through the Pump Seal (PHTS)	(3)		
15.7.2.2	Inadvertent Release of Oil through the Pump Seal (IHTS)	(3)		

TABLE 15.3-3 (Continued)

Events		Required Operable System	Primary	Secondary
15.7.2.3	Generator Breaker Failure to Open at Turbine Trip	(3)		
15.7.2.4	Rupture of RAPS Surge-Vessel	(3)		
15.7.2.5	Liquid Radwaste System Failure Tank	(3)		
15.7.3	Extremely Unlikely Events			
15.7.3.1	Leak in Core Component Pot	+EVTM seals		
15.7.3.2	Spent Fuel Shipping Cask Drop from Maximum Possible Height	(3)		
15.7.3.3	Maximum Possible Conventional Fires, Flood, Storms or Minimum River Level	(3)		
15.7.3.4	Failure of Plug Seals and Annuli	Containment Isolation System		
15.7.3.5	Fuel Rod Leakage Combined with IHX and Steam Generator Leakage	(3)		

Notes to Table 15.1.3-3

(1) The redundant and diverse decay heat removal capabilities are described in Section 5.6.

(2) The plant controller is designed to bring the plant to a steady state operating condition in the event of turbine trip. Reactor shutdown may be initiated by the operator if desired. PPS would react (Primary-Steam Feedwater, Secondary-Evaporator Outlet Temp.) only if the (non-safety class) plant controller failed.

(3) These events do not constitute challenges requiring operation of plant protection systems to protect the health and safety of the public.

(4) Where more than one trip function is listed, the separate trip functions would each adequately mitigate the event, though the first trip listed is expected to act first to safely shutdown the plant. 22

15.1.4 Effect of Design Changes on Analyses of Accident Events

The design of the CRBRP has made significant progress since the consequences of design basis events reported in the remainder of this chapter were first analyzed. A review of approved design changes to determine which may affect the reported results and a qualitative evaluation of the effects of these changes has been made. A primary example is the change in core design from a homogeneous to a heterogeneous configuration. The results of this effort are discussed in the following sections.

15.1.4.1 Reactivity Insertion Design Events

Section 15.2 covers the analyses of reactivity insertion design events. The format progresses from anticipated up through faulted design transients with each accident scenario providing:

- identification of causes and accident description;
- analysis of effects and consequences;
- conclusions.

With regard to accident scenarios, there have been no changes to Section 15.2 since the original PSAR submittal. However, various pieces of design data have changed and have subsequently been incorporated into the appropriate design sections of the PSAR. Modifications to the nuclear and thermal-hydraulic information affect the maximum temperatures attained and the temperature/time traces shown. The purpose of this section is to indicate the effect of these various changes to the Section 15.2 results.

Reactivity insertion accidents typically result in overpower transients that are characterized by an increase in power such that a proportionately larger increase occurs in fuel temperature than in cladding temperature. This is opposed to undercooling design events which have a very small fuel temperature increase as compared to that of the cladding. Worst case overpower conditions commonly have a rapid increase in power which institutes scram of the Plant Protection System (PPS). For events having a rapid power burst, the period of the overpower conditions is typically less than one second (see Figure 15.2.3.3-3, for example). Although the shutdown occurs quickly, effects such as fuel melting and the potential for fuel/cladding interaction are of prime importance in the fuel pin performance evaluations.

To demonstrate temperatures that envelope overpower events with current data applied, a worst case event was reanalyzed and the results are herein described relative to the former values. This worst case selected previously was the Seismic Reactivity Insertion (SSE) (see Section 15.2.3.3.1) with primary control system shutdown (which is an extremely unlikely event).

Secondary shutdown has been selected for this analysis because of the longer scram delay time noted below. As with the past analyses, the following conservative assumptions were made:

1) All full power cases are for the reactor operating at thermal hydraulic design conditions with a power generation of 975 MWt at three-loop operation. (Power uncertainties are discussed in Section 4.4.3.2.)

2) Since the smallest Doppler coefficient occurs at the beginning-of-equilibrium cycle, the transient reactor power calculation was made for this particular phase in core life. This results in the highest possible reactor power changes being calculated. Table 4.3-16 in Section 4.3.2.3 gives the total nominal Doppler constant of 0.002697 at BOC3 used for fuel plus axial blankets. This value is then reduced 30% to account for 3σ uncertainties.

3) The highest cladding and fuel temperatures for fuel assembly hot rods (in F/A #52 and 101, respectively) occur at the beginning of the first cycle of operation. The conservative reactor power calculation from Item 2 above was applied to these particular rods. With burnup, the power generation and steady state temperatures decrease (flows are constant) in the hottest fuel assemblies, and consequently the temperature due to the transients would decrease after beginning of cycle. Since the highest cladding temperature for the blanket rods occurs at end-of-life, this condition is conservatively analyzed using the item 2 power for blanket hot rod analyses.

4) As described in Section 7.2.1.2 and Section 4.2.3.1.3, the maximum allowable time delays for PPS logic and electrical/machanical delays have been conservatively enveloped by using a 200 millisecond delay between the instrument channel output going beyond the trip level and start of control rod insertion.* The trip signal for the ensuing analysis is from the loss of power trip function on the secondary control rod system which has a maximum delay of 0.8 seconds from the time power is lost to the primary pumps.

5) Worst case, required rates of shutdown worth used for control system scram reactivity assuming highest worth rod stuck (described in Section 4.2.3.1.3).

6) Worst combination of timing sequence between the loss-of-power to the primary pumps and the step reactivity insertion to the core (as described in Section 15.2.3.3.1).

7) Three sigma (3) hot channel factors were used for the analyses and the cladding temperatures shown are the inner surface of the cladding at the highest temperature position, both axially and circumferentially on the rods (position is under the wire wrap).

*In this instance the sensor delay has been encompassed by the 200 msec PPS logic and control rod unlatch delay. This is justified by the small magnitude of the flux sensor delay which is estimated at less than 10 msecs.

Results from FORE-2M analysis are given in Figures 15.1.4-1, 2 and 3 and Table 15.1.4-1 for a 60¢ step reactivity insertion occuring at the worst time during the SSE (see Section 15.2.3.3.1). Comparisons of the heterogeneous core results are made with data for a homogeneous core previously reported in this section. This previously reported data updated earlier data for the homogeneous core analyzed in Section 15.2.3.3. The figures show the maximum hot rod temperatures for F/A #52 and 101 in the heterogeneous core as compared to similar data for F/A #6 and #8 in the homogeneous core. In the homogeneous core, F/A #8 had the maximum cladding temperature hot rod and F/A #6 had the maximum fuel temperature hot rod. In the heterogeneous core, F/A #101 has the maximum fuel temperature and F/A #52 has the maximum cladding temperature. A comparison of pertinent parameters is given by Table 15.1.4-1. Although the temperatures for the heterogeneous core are somewhat higher, the results are well within the limits given by Table 15.1.2-2 (no sodium boiling) for extremely unlikely faults. In addition, the maximum coolant temperature is less than 1600°F. This provides considerable margin to the coolant saturation temperature which is greater than 1800°F when this maximum is attained.

One important note with regard to the above comparison is that credit has been taken in the heterogeneous core hot rod evaluation for having a programmed startup to enhance the power-to-melt (i.e., improved fuel restructuring and gap conductance) by using LIFE-III analyses. The homogeneous core studies used fuel rod conditions calculated using P-19 data which assumes a "fresh rod" with no previous operation. Section 4.4.3.3 of the PSAR addresses how programmed startup is achieved and the consequential improvement on the minimum power-to-melt. Since the proposed programmed startup is still not optimized, further improvement can be achieved through optimization of the programmed startup which is scheduled for the FSAR as discussed in Section 4.4.3.3.

Similar data to that described above for the hot fuel assembly rod is not available for the hottest blanket assembly rod. However, it has been found from past analyses shown in Section 15.2, that the temperatures for the blanket size rod are significantly less for the SSE step reactivity insertion type of event. This is due to the large thermal inertia of the significantly bigger blanket size rods. With the extremely quick power rise for the 60¢ step SSE case, there is insufficient time for the temperatures to increase as much as they do for the smaller fuel size rods. Thus, the fuel hot rod temperature response represents the worst case thermal transient effect to the core. These results support a qualitative conclusion that the design changes incurred since the original PSAR submittal are not expected to significantly change the results reported in Section 15.2.

15.1.4.2 Undercooling Design Events

Section 15.3 covers the analysis of undercooling design events. Subsequent to the earlier analyses, various design changes to the plant have taken place. The most important design change is, of course, the core design change to the heterogeneous core configuration.

An impact assessment of the significant design changes as discussed below indicated that their effects on the consequences of undercooling events are either positive or insignificant. Nevertheless, to positively demonstrate the adequacy of the current plant design configuration against undercooling events, a detailed re-analysis was undertaken. This re-analysis was based on a 'worst-case' event selected in a systematic manner by reviewing the earlier results of the undercooling event analyses reported in Section 15.3. Unlike the overpower transients, undercooling events typically involve very small fuel temperature increases as compared to that of the cladding. The selection of the 'worst-case' event was therefore primarily based on the worst consequence in terms of maximum cladding temperatures. The 'worst-case' undercooling event so established was then re-analyzed consistent with the current plant design. Details of and the results from this re-analysis are provided herein following an item-by-item discussion of design changes below.

The significant design changes with respect to the undercooling events and their expected effects are:

1. The heterogeneous reactor core arrangement is described in Chapter 4. Although the heterogeneous core arrangement is substantially different from the homogeneous arrangement, the nuclear and thermal-hydraulic design constraints are very similar. Since no design basis undercooling event presented a significant challenge to the homogeneous core (analysis indicated a margin of approximately 200°F to the onset of sodium boiling for the homogeneous arrangement), the same is predicted to hold true for the heterogeneous arrangement.

2. A minimum flow coastdown requirement has been specified for the primary sodium coolant pumps leading to a corresponding minimum pump-and-drive system inertia requirement which is larger than the value used in the PSAR Section 15.3 analyses. This also applies to the intermediate sodium coolant pumps which use an identical design. The impact of the increased net primary flow during the coastdown is in the direction of less severe consequences for undercooling events through additional heat removal.

3. Refinements made in primary sodium piping layouts which either change flow resistance or transport time are expected to have minimal, if any, effect on the undercooling event.

4. The differences in IHTS piping configuration between that considered for Section 15.3 analyses and the present design are expected to have minimal effects on the undercooling transients of Section 15.3. The changed transport delay further increases transport time for steam-system induced transients to reach the reactor inlet.

5. The current primary system cold leg check valve design is expected to allow limited reverse-flow leakage in a loop in which the pony motor is not operating. An analysis has been performed using the DEMO Code which shows that adequate core decay heat removal can be maintained with only a single primary pony motor operating, permitting reversed-flow in the other two loops. Under these conditions, loop thermal heads effectively limited the reversed flow well within the check valve leakage specification and, in fact, maintained a small forward flow during most of the transient duration. Consequently, adequate safety performance is expected for events in which check valve leakage is a factor.

6. The IHX design has been changed from a removable tube bundle to a fixed tube bundle design. This change does not affect the significant IHX thermal and hydraulic parameters used in the Section 15.3 analyses.

7. A length increase has been made in the steam generator modules since completion of the PSAR Section 15.3 transient analyses. Neither water-side nor sodium-side pressure drop changes are large. The module surface increase is in the direction to decrease severity of undercooling transients.

8. Piping layouts have been optimized and simplified on the water/steam side. Two drum headers have been eliminated and replaced by direct-to-drum pipe connections, in conjunction with the use of an annular-inlet girth-baffle in the drum (see item 9 below). Available elevation difference to assist recirculation flow has been increased (drum raised 7.0 feet).

   The larger-diameter nozzles at the drum increase the available discharge area in case of large steam pipe breaks immediately at the drum. It should be noted, however, that in general, effects of severe steam-side events reach the reactor well after the reactor has been shut down by the Plant Protection System because of the extended transport time at pony-motor flow rates. Consequently, no safety problem results at the reactor.

9. As noted above, the steam drum has been redesigned using an integral annular girth baffle at the inlets from the evaporator, with elimination of the steam inlet and exit headers (the recirculation exit header remains). The possible effect on discharge area available during a steam pipe break were noted above. Increased drum elevation is expected to increase water recirculation for cases in which the recirculation pump is not operating. (See pertinent comments in Item 8 immediately above.)

10. A revised head curve for the steam system recirculation pump is now available. Comparison of these revised characteristics with those used in the Section 15.3 analyses indicate that for steam system transients where the recirculation pump head has an effect on results, the former characteristics are more conservative. (Consequently, design transient analyses have continued to use the prior characteristics.)

11. The Plant Control System design has progressed significantly since the Section 15.3 analyses were performed but the control concept has not changed. For the analyses of Section 15.3, however, early trip action by the Plant Protection System precluded Control System effects on the transient results (I and C tolerances and deadband are reflected in the conservative starting conditions for the transient analyses). The Plant Protection System functions having an action during the Section 15.3 events are tabulated in Table 15.1.3-1. The functions given remain valid, with the exception that the 30% trip mismatch setting (offset) on flux-to-flow may be reduced to accommodate events leading to increases in power-to-flow ratio that might level-off just before the trip level. The effect of this reduction will be to decrease time required to trip and thus will lessen the effects of the resulting transient. In addition, a requirement has been established for the secondary rod system to generate a Trip Signal within 0.8 seconds after loss of electrical power on the pump busses.

12. Two Control and Protection System developments for Balance of Plant having a potential bearing on the Section 15.3 transients are: (a) a better definition of the feedwater controls and (b) incorporation of a delayed turbine trip following reactor plant trip. The latter is used to reduce steam pressures immediately following a plant trip to prevent safety-valve operation. These changes will not affect reactor conditions immediately after shutdown, because of the extended sodium transport times in the PHTS and IHTS at pony motor flowrates.

13. In the auxiliary feed system, regenerative heating of the auxiliary feedwater has been removed, with the result that cold auxiliary feedwater is now injected. For cases in which recirculation pump power is not available, mixing of the drum water with the cold feedwater may be reduced through stratification effects, resulting in lower transient cold-leg temperatures.

The re-analysis of a worst-case event is discussed below. The worst-case event selected was the Loss of Offsite Electrical Power (Section 15.3.1.1) as it resulted in the worst maximum cladding temperature. The re-analysis was performed with DEMO-4 and FORE-2M that have incorporated all the design changes discussed above. The same PPS design data and conservative approach as described in Section 15.1.4.1 were also used for this analysis.

Parametric studies were performed with the earlier version of FORE-2M (Reference 1) to substantiate a worst-case combination for the selected event for the nuclear power variation calculation. The objective was to seek a set of conditions that leads to the highest power (or the slowest decrease in power) during the transients. Conservatively, this model used a reduced level of core detail (from the 7 radial by 7 axial mode capability of the code) where the total core wide Doppler for the fuel and blanket regions was included in the fuel region. A base case nuclear model was established using the following conditions:

- o Minimum C_p of fuel;
- o Longest flow coastdown of the primary pumps;
- o Zero decay heat;
- o Maximum fuel/cladding gap conductance;
- o Zero sodium coolant density feedback; and
- o Maximum Doppler coefficient

All other feedbacks which are negative (such as fuel expansion, cladding expansion, core radial expansion and bowing) were conservatively neglected. Results of this base case and the effect of each worst case parameter are given by Table 15.1.4-3. Here the variation in the neutronic power is shown for each significant parameter. This demonstrates the importance of each condition in establishing the base case. In addition to these conservative studies, the base case was repeated using the current FORE-2M capability which considers all core regions. With the same base case conditions, a maximum neutronic power variation of 0.1276* was found as compared to the value of 0.1327* in Table 15.1.4-3. Likewise, a nominal data case was run with this model and a maximum neutronic power variation of 0.1265* resulted. This confirms the conservatism of the base case model.

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*Value quoted at 2 seconds into transient for comparative purposes.

Although the above nuclear base case conditions were used for evaluating the neutronic power variation, they do not necessarily form the worst case condition for the hot rod temperature calculations. For this type analysis, conditions like the quickest flow coastdown provide the worst case. A mismatch of conditions (between the neutronic and temperature calculations) was thus conservatively selected to calculate worst case transient hot rod temperatures.

The results of the re-analysis are summarized on Figures 15.1.4-7 and 8 for the worst case fuel and blanket assemblies, respectively. The maximum fuel rod hot spot cladding temperature (3σ) in the heterogeneous core hot channel (as given in Figure 15.1.4-7) is 1455°F for F/A #52 (see Figure 4.2-10B for location). This compares to 1500°F fuel rod hot spot cladding temperature superceded the 1630°F result discussed in Section 15.3.1.1. Maximum cladding temperatures for the inner and radial blanket assemblies can be seen to be 1471°F and 1478°F, respectively, from Figure 15.1.4-8. These temperatures for both the fuel and blanket assemblies are well within the applicable Fuel Design Limits (Table 4.2-35 and Table 15.1.2-2).

To demonstrate the conservatism of the above 3σ hot channel analyses, the F/A #52 case was reanalyzed using nominal data. For this case, the maximum cladding temperature reached was more than 175°F lower relative to the worst case prediction as shown by Figure 15.1.4-7.

From the large margin as demonstrated by the results from the detailed re-analysis discussed above, it can be seen that the heterogeneous core can adequately accommodate all Undercooling Events described in Section 15.3.

15.1.4.3 Local Failure Events

15.1.4.3.1 Introduction

Section 15.4 describes the basis for the position that rod-to-rod failure propagation will not occur for faults in the fuel, radial blanket and control assemblies of CRBRP. The evaluation used enveloping values for the various input parameters and other attendant conservative assumptions. The results contained in Section 15.4 indicate that there are substantial margins available to prevent failure propagation from local faults.

The analysis covered a whole spectrum of local fault initiator events, including: stochastic failure, localized (fuel pin) overpower, flow blockage, and small gas bubble passing-through. The core design change from the homogeneous core to the heterogeneous core resulted in various changes to the input parameters used in the earlier analyses. These changes, however, largely led to various degrees of improvement over the calculated margins obtained earlier. Nevertheless, detailed re-analyses were undertaken for two reasons: 1) to positively confirm that the heterogeneous core has adequate margin against local fault events; and 2) there had been no analysis for Inner Blanket Assemblies that did not exist in the homogeneous core configuration.

In view of the fact that changes to input parameters due to the core design change largely create positive impacts, the reanalyses were done by selecting in a systematic manner an 'enveloping' initiator event. The 'enveloping event' was then analyzed in detail for a given type of core assemblies. A summary of the basis used and considerations involved in the 'enveloping' event selections is provided below.

Key parameter changes due to the core design change were compiled. Each was then assessed for impact on: probability of occurrence, potential for failure propagation, and severity of consequence of each local fault initiator event. If the impact is relative only to a single initiator event or to a particular type of assembly, this is also noted. A brief discussion of these key parameter changes is as follows.

- o <u>Lower Fission Gas Pressure</u> Due to the reduced burnup in the heterogeneous core fuel management scheme relative to the homogeneous core fuel management scheme (2 versus 3 cycles residence time) and refined methods of calculations, the fission gas plenum pressure is substantially lower in the fuel rods (approximately 1000 psi vs. 1710 psi). This reduces the potential for occurrence of stochastic fuel pin failures and enhances fuel rod cladding rupture margin against local fault events across the board, either due to overpower or undercooling. Further, it also tends to reduce the potential for propagation of a local fault when one occurs.

  The fission gas plenum pressure in the heterogeneous core blanket assemblies is also lower (250 psi and 280 psi in the inner and radial blanket assemblies, respectively, vs. 380 psi in the homogeneous core blanket assemblies). Also, in the control assemblies, the helium gas pressure is substantially lower. Based on preliminary analysis results (Table 4.2-46), the gas pressure is about 1037 psi vs. 3352 psi in the homogeneous core.

- o <u>Single Enrichment</u> The heterogeneous core has a single enrichment. This eliminated the potential for localized overpower faults due to misloading of a fuel assembly in a wrong enrichment zone.

- o <u>Less Positive Void Worth in Fuel Assemblies</u> The bulk sodium void worth for the fuel assemblies is less positive in the heterogeneous core (2.31 \$ vs. 4.00 \$ for the whole core). This improves the margin in fuel assemblies against local faults due to small bubble passing-through.

- o <u>No Radial Blanekt Assemblies Shuffling</u> Based on the present fuel management scheme, there will be no shuffling of radial blanket assemblies. This eliminated the potential for misloading radial blanket (RB) assemblies at a wrong design position in the core and thus the attendant local faults in RB assemblies due to localized overpower.

- <u>Lower Maximum Linear Power in RB Assemblies</u> The (3σ) maximum linear power rating in the RB assemblies is much lower (14.5 kw/ft vs. 18.3 kw/ft). This reduces the potential for failure propagation due to local flow blockage in RB assemblies. It also enhances the power-to-melt margin in RB rods and thus helps ameliorate the potential consequences of a local fault due to localized overpower. The maximum linear power rating of IB/A is about the same (18.4 kw/ft).

- <u>Higher Maximum Linear Power in Fuel Assemblies</u> The (3σ) maximum linear power rating of the fuel assemblies is higher in the heterogeneous core (14.1 kw/ft vs 12.8 kw/ft) and the peak assembly average linear power rating is also higher (9.5 kw/ft vs 8.4 kw/ft). This primarily tends to impact local faults due to flow blockage in a fuel assembly. Relative to the homogeneous core configuration, it may negatively affect the thermal consequences and the potential for propagation of a local fault as a result of coolant subchannel blockage.

- <u>Higher Mean Outlet temperature in IB/A</u> The (3σ) maximum outlet temperature of the heterogeneous core inner blanket assemblies at THDV is 1096°F (vs. 1026°F in blanket assemblies in the homogeneous core) and the peak assembly average linear power is also higher (11.3 kw/ft vs. 3.7 kw/ft). This primarily impacts the propagation potential for and thermal consequences of local flow blockage faults relative to the homogeneous core analysis for blanket assemblies.

  Although the peak assembly average linear power rating in the radial blanket assemblies of the heterogeneous core is also relatively higher (6.8 kw/ft vs 3.7 kw/ft), the peak linear power rating is lower and the power-to-melt margin is greater as discussed above. Accordingly, the net impact is relatively insignificant for the radial blanket assemblies.

- <u>Higher Plutonium Content</u> The Pu content in the heterogeneous core fuel is higher. This tends to affect the fuel-coolant reaction product formation and thus the failure propagation of failed fuel. However, any such impact only applies to long-term operation with failed fuel. Further, the required low level of oxygen concentration in the primary sodium makes the formation of fuel-coolant reaction products a very slow process. Therefore any effect of the higher Pu content on fuel element failure propagation is expected to be insignificant.

As can be seen from the above discussions, the core design change largely led to positive impacts on the margin against local fault events. A few negative impacts that could be of some significance were identified. All of them, however, primarily relate to the local faults of flow blockage. The systematic detailed parametric study for selecting 'enveloping' events led to the same conclusion. The 'local flow blockage' fault was thus selected for detailed re-analyses for the fuel assemblies and the inner blanket assemblies. The core design change led to predominantly positive impacts on margins against local fault events so far as the radial blanket and control assemblies are concerned. Therefore, no re-analyses were undertaken for these two types of assemblies.

15.1.4.3.2 Fuel Assemblies

The same worst-case local flow blockage fault analyzed in detail earlier (Section 15.4.1.3.3) was re-analyzed for the heterogeneous core fuel assemblies. The same input assumptions pertaining to the accident scenario and the heterogeneous core design parameter values were used for the re-analysis. The results are delineated in Tables 15.1.4-4.

As can be seen from the results shown, the maximum coolant temperature in the wake region and the maximum fuel rod cladding temperature are both lower for the heterogeneous core (1216°F vs 1261°F and 1367°F vs. 1396°F, respectively). The same temperatures for the hot pin are slightly higher but are still well below the temperature corresponding to prompt cladding failure as discussed in 15.4.1.3.3.

15.1.4.3.3 Inner Blanket Assemblies

The same enveloping local fault event was analyzed in detail for the inner blanket assemblies. The IB/A design parameter values were used as the input for this analysis. As discussed in Section 15.4.3.3.1, flow blockages of blanket assemblies are extremely unlikely. Nevertheless, the same conservative scenario of a six-channel blockage was postulated for the detailed analysis. In view of the fact that the total number of subchannels is much smaller for the blanket assemblies, six channels in a blanket assembly amount to a much greater percentage of the total assembly coolant flow area.

Table 15.1.4-5 delineates the results from this analysis. The maximum coolant temperatures in the wake region for both the peak pin and hot pin are considerably lower than the estimated saturation temperatures in both cases. The maximum cladding temperatures are also within the acceptable limits.

15.1.4.3.4 Radial Blanket Assemblies

The core design change to the heterogeneous core creates predominantly positive effects on the radial blanket assemblies and further improves the margin against failure propagation due to local faults.

The fission gas plenum pressure, both an initiator for stochastic faults and a driving force for other types of local faults, is about 24% lower in the heterogeneous radial blanket assemblies (280 psi vs. 380 psi in the homogeneous core).

The maximum linear power rating is lower by more than 20% (14.5 kw/ft vs. 18.3 kw/ft). This provides further improvement over the margins against essentially all the types of local faults. The radial blanket assemblies in the heterogeneous core are all located in the outer rings. The power-to-melt margin is increased although the peak assembly average linear power is higher and this enhances the margin against localized overpower faults.

There is no shuffling of radial blanket assemblies in the heterogeneous core. This essentially eliminated the potential for loading a RB/A in a wrong position and the attendant local faults in RB/A due to localized overpower.

From the discussions above, it is clear that the radial blanket assemblies in the heterogeneous core have actually greater margin against fuel element failure propagation due to local faults than that demonstrated by the earlier analyses and discussed in Section 15.4.3.

15.1.4.3.5 Control Assemblies

All the input parameter values used in the earlier analyses as discussed in Section 15.4.2 remain essentially unchanged, although the peak linear power and the peak assembly average linear power are both slightly lower than the homogeneous core. The only significant change relates to the peak (helium) gas plenum pressure. The peak gas plenum pressure of the heterogeneous core control assembly rods is substantially lower than that of the homogeneous core (1037 psi vs. 3352 psi). This provides significant improvement over the margin against local failure events. Therefore, the heterogeneous core control assemblies clearly have been greater margin against local failure propagation than that demonstrated earlier as discussed in Section 15.4.2.

Based on the assessment discussed above and the results of the detailed re-analyses provided, the heterogeneous core can be seen to have been more margin for local faults than the homogeneous core in some cases or substantially the same margin in others.

15.1.4.4 Effect of Design Changes on Radiological Consequences

The changes associated with the heterogeneous core (decrease in the number of fuel assemblies, decrease in the total energy produced in the maximum power fuel assembly during its life, increase in the total core plutonium loading, and rearrangement of the fuel and blanket assemblies) is expected to have no significant effect on the radiological consequences of the accidents reported in Sections 15.5 through 15.7. For the accidents considered, the consequences are dependent upon the total radioactivity contained within a single fuel assembly, the radioactivity contained in the sodium coolant, or the radioactivity released to the cover gas. These are discussed below.

The consequences of accidents involving single fuel assemblies are controlled by the inventory of noble gas and volatile fission products. These in turn may be classed as short lived (half-life less than a few months) and long lived (half-life greater than a few years). The inventory of short lived radioactivity will be dependent upon the power level in the assembly which, for conservatism, is assumed to be the assembly producing the maximum power. The increase in power level for the maximum power assembly, which resulted from the changes in the core, was approximately 3.4 percent. The long lived radioactivity will be dependent on the total energy produced by the assembly during its life. For the maximum-power assembly, this is expected to be about 80 percent of that for the homogeneous core, as reported in subsequent sections of Chapter 15. Therefore, the presently reported doses for accidents involving a single fuel assembly are not expected to increase by more than 3.4

percent as a result of the changes made to the reactor core. Accidents involving single fuel assemblies are discussed in Section 15.5.2.1, 15.5.2.2 and 15.5.2.3 and resulting doses for the more severe accident reported in Table 15.5.2.3-4.

Considering that the margins available are greater than two orders of magnitude, the small increase in dose due to the design change will not affect the conclusions reached concerning the safeguards provided for the accidents.

The consequences of accidents involving spills of sodium and subsequent fires are directly proportional to the radioactivity contained within the sodium. This, in turn, is dependent upon the failure rate of fuel pins. The design limit of failure of fuel pins producing one percent of the core power has not been changed and consequently the fission product activity within primary sodium coolant will not increase. The inventory of long lived fission products may actually decrease due to the shorter residence time of fuel assemblies within the core. The limit of 100 ppb of plutonium within the primary sodium coolant will also be retained. The design limit on fuel pin failure will also maintain the activity on the sodium clean up subsystems within the same design envelope. No increase in the consequences of accidents involving the primary coolant are expected as a result of the changes to the core design.

The activity contained within the reactor vessel cover gas is directly proportional to the failure rate of the fuel pins. As indicated above, the design limit for this occurrence will not change. The cover gas clean-up systems will also handle the same inventory of radioactivity (except that the long lived activity may be reduced). Therefore, no increase in the consequences of accidents involving the reactor vessel cover gas are expected as a result of the changes in the core design.

15.1.4.5 Reactor Assembly Bowing Reactivity Considerations

As indicated in Section 4.2.2.4.1.8, reactor assembly bowing can cause positive reactivity to be inserted during variations in the portion of the power/flow ratio range below 0.7. Accordingly, reactor transient events considered in Section 15.2 were reviewed to determine the significance of bowing reactivity additions on the progress and consequences of these events. The following conclusions were reached:

15.1.4.5.1 Reactor Startup Transient (0-40% Power)

Quantitative analyses were performed for the 2 /second rod withdrawal transient initiating in the startup range at 8% power, 40% flow (worst case initial condition) incorporating the bowing reactivity characteristics described in Section 4.2. The results of this analysis indicate that the thermal consequences on the fuel and cladding for this event are less severe than if the reactivity effect of bowing had been neglected. This is because positive bowing reactivity addition causes power to rise more rapidly. Thus, the reactor trip occurs at an earlier point in the time of the event before significant changes in temperature have occurred. Consequently, the effect of adding positive reactivity due to reactor assembly bowing is analogous to the addition of larger ramp reactivity insertions, e. g. 5 /second versus 2 /second. As indicated in Section 15.2, the faster ramp rate causes a reactor trip to occur earlier in the event and peak cladding and fuel temperatures during the event are lower.

For events with large reactivity insertions (e. g., 30 for OBE or 60 for SSE), a reactor trip would occur prior to any significant reactor assembly bowing reactivity addition. The power increases to trip points much quicker than significant temperature changes can occur in the duct structure.

15.1.4.5.2 <u>Reactor Power Range Transient (0-100% Power)</u>

For these events, reactor assembly bowing results in a negative reactivity effect. By neglecting the feedback of reactor assembly bowing, a conservative determination of the reactor temperature is obtained for these events.

TABLE 15.1.4-1

COMPARISON OF THE HETEROGENEOUS AND HOMOGENEOUS CORE

HOT ROD TEMPERATURES FOR 60¢ STEP REACTIVITY INSERTION UNDER SSE**

DESIGN	MAXIMUM REACTOR POWER INCREASE (P/Po)	MAXIMUM CLADDING TEMPERATURE (°F)	MAXIMUM FUEL TEMPERATURE (°F)
Homogeneous			
F/A #8	2.47	1578	4757
F/A #6	2.47	1555	4954
Heterogeneous			
F/A #52	2.61	1647	4815
F/A #101	2.61	1625	5041*

*Approximately only 4.8% of the pellet cross-section at the center of active core would have a calculated temperature greater than 5000°F

**This table provides analysis results assuming primary RSS actuation. The effects of secondary RSS actuation only (no primary RSS actuation), including conservatively predicted fission gas release, are discussed in response to NRC Question CS490.23.

NOTE: Po is initial steady-state operating power.

TABLE 15.1.4-2 has been intentionally deleted.

TABLE 15.1.4-3

PARAMETRIC CASES TO DETERMINE WORST CASE FOR NEUTRONIC POWER VARIATION DURING THE LOSS OF OFF-SITE ELECTRICAL POWER EVENT

CASE	CONDITIONS	$P/P_{o,n}$*
1 (Base Case)	o Minimum fuel C_p o Longest flow coastdown o Maximum Doppler o Zero decay heat o Maximum fuel/cladding gap conductance o Zero sodium coolant density feedback	0.1327
2	o Quickest flow coastdown o Other conditions same as Case 1	0.1322
3	o Maximum sodium coolant density feedback o Other conditions same as Case 1	0.1326
4	o Maximum decay heat o Other conditions same as Case 1	0.1322
5	o Maximum fuel C_p o Other conditions same as Case 1	0.1325
6	o Minimum fuel/cladding gap conductance o Other conditions same as Case 1	0.1304
7	o Minimum Doppler o Other conditions same as Case 1	0.1248

*P = Nuetronic power at 2 seconds (selected for comparison only) into the transient.

$P_{o,n}$ = Neutronic power at time 0 of the transient.

TABLE 15.1.4-4

TEMPERATURES BEHIND A CENTRAL SIX CHANNEL BLOCKAGE IN FUEL ASSEMBLIES

	PEAK PIN[3]	HOT PIN[4]
Maximum wake temperature increase, ^{o}F	270	116
Average wake temperature increase, ^{o}F	180	77
Maximum wake temperature, ^{o}F	1216	1385
Maximum cladding temperature, ^{o}F[1]	1367	1432
Dimensionless residence time, t_R[2]	22.5	22.5
Linear power rating, Kw/ft	14.1	4.76

---

(1) Based on maximum fluid temperature.

(2) $t_R = d_B \frac{\tau U}{}$ where: τ is the average residence time of the fluid in the wake region, U is the free stream velocity; and d_B is the characteristic blockage dimension. The temperature increase is proportional to t_R. The t_R value 22.5 used here is conservative.

(3) Blockage is conservatively assumed to occur at peak power spot, i.e., core Midplane.

(4) Blockage is conservatively assumed to occur at hot spot, i.e., top of the core.

TABLE 15.1.4-5

TEMPERATURES BEHIND A SIX-CHANNEL BLOCKAGE IN INNER BLANKET ASSEMBLIES

	PEAK PIN(3)	HOT PIN(4)
Maximum wake temperature increase, ^oF	462	167
Average wake temperature increase, ^oF	308	111
Maximum wake temperature, ^oF	1410	1385
Maximum cladding temperature, ^oF(1)	1526	1430
Linear power rating, KW/ft	18.4	6.7
Dimensionless residence time, t_R(2)	22.5	22.5

(1) Based on maximum fluid temperature.

(2) The value of t_R was calculated to be 17 (Section 15.4.3.3.3). For conservatism, the same t_R value 22.5 is used here.

(3) Blockage assumed to occur at core Midplane.

(4) Blockage assumed to occur at top of the core.

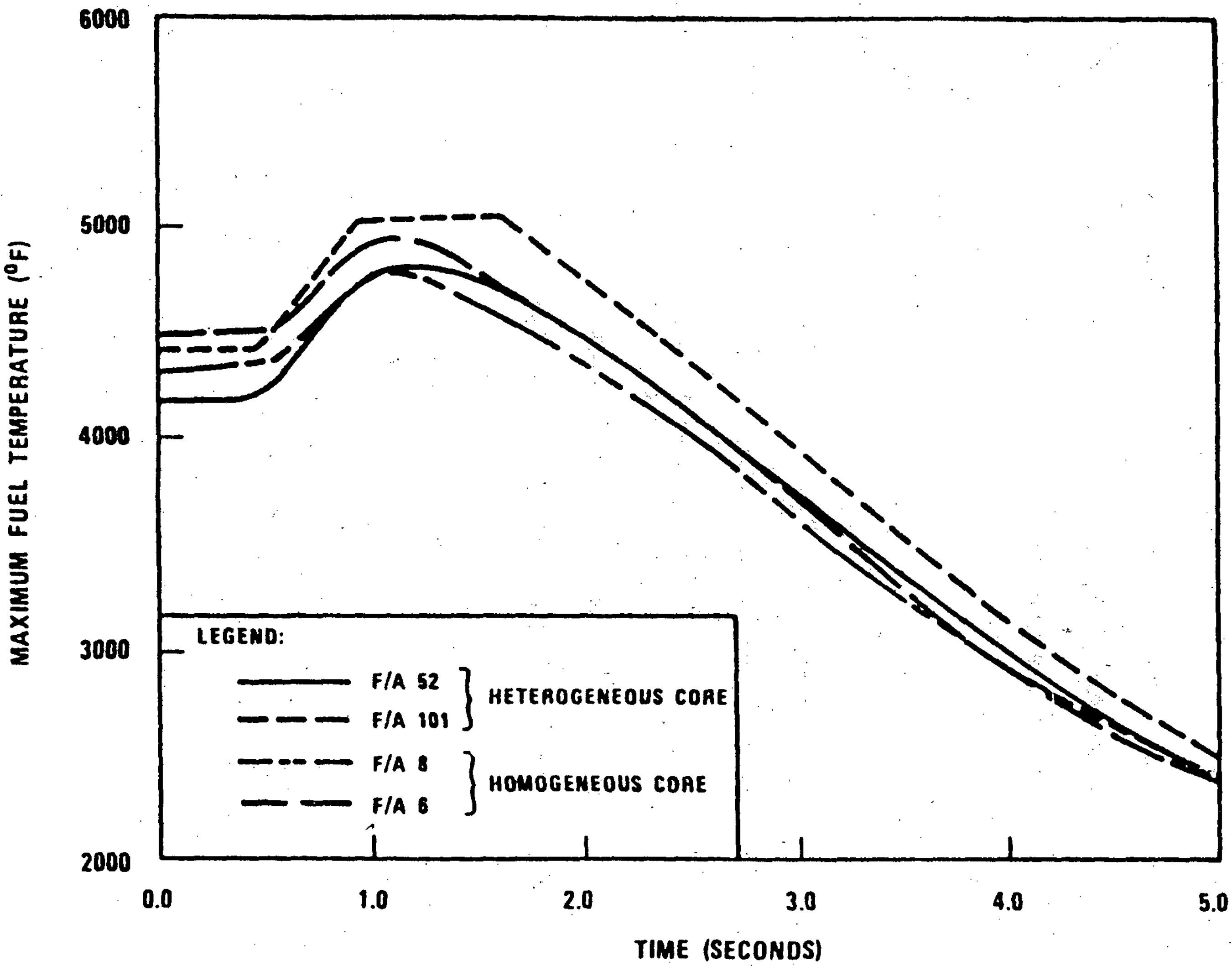

Figure 15.1.4-1 Variation of F/A Hot Channel Maximum Fuel Temperature for a 60¢ SSE Transient

1764-62

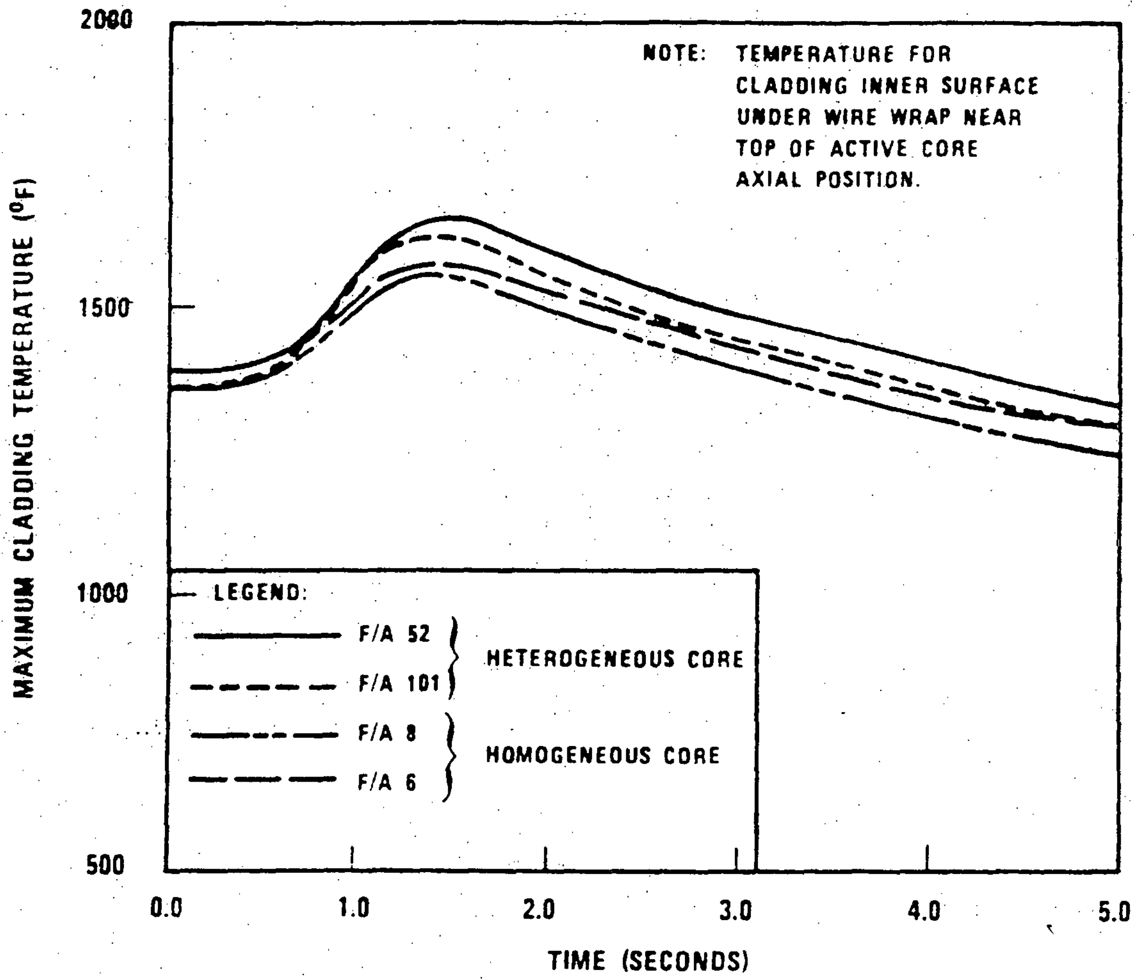

Figure 15.1.4-2 Variation of F/A Hot Channel Maximum Cladding Temperature for a 60¢ SSE Transient

1764-63

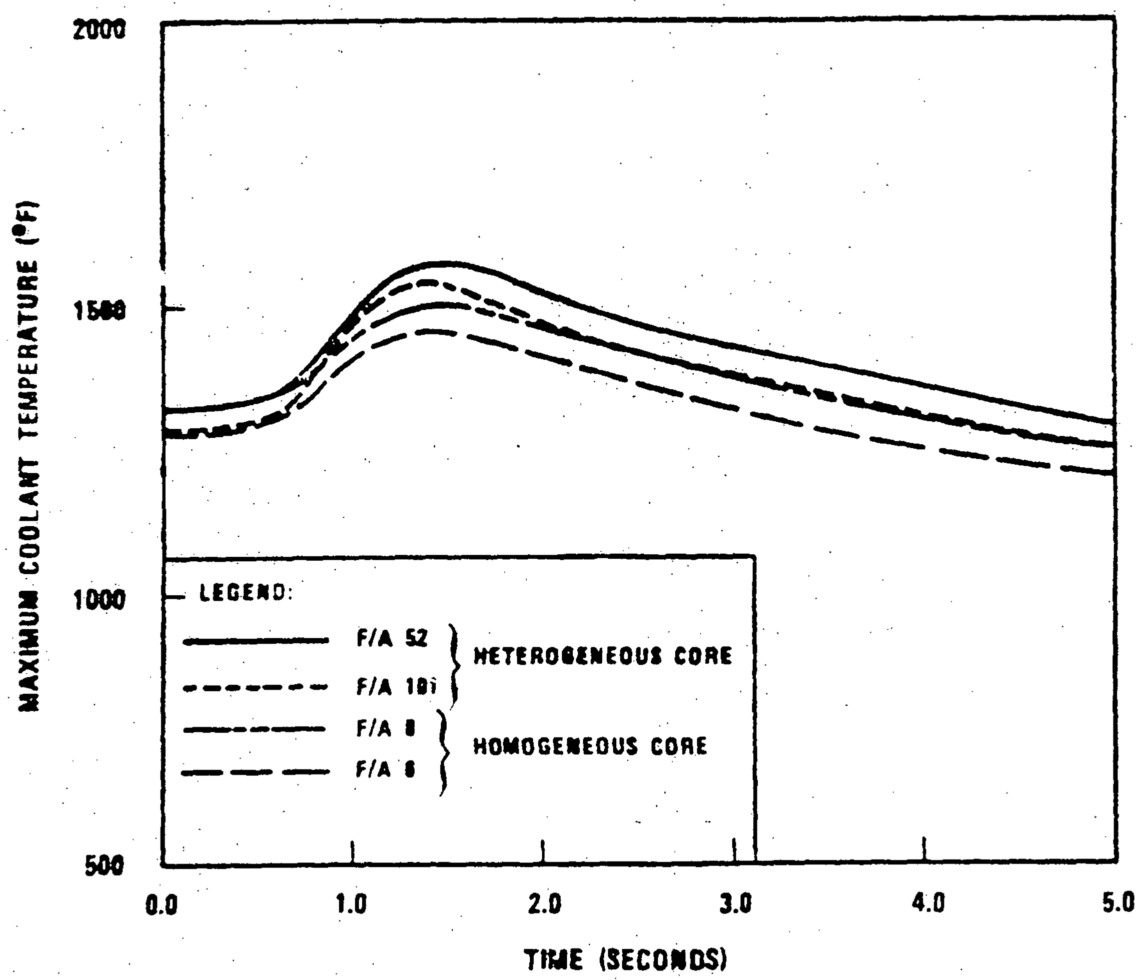

Figure 15.1.4-3 Variation of F/A Hot Channel Maximum Coolant Temperature for a 60¢ SSE Transient

1764-64

FIGURES 15.1.4-4 through 6 have been intentionally deleted.

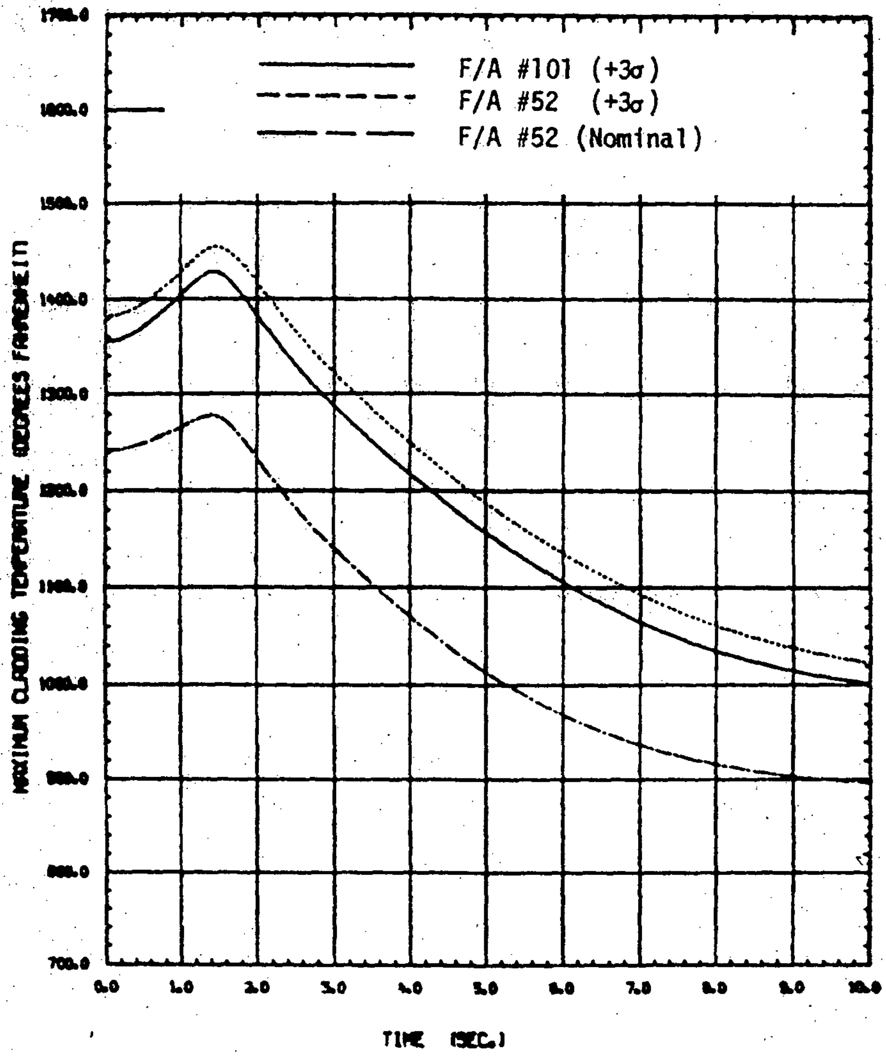

FIGURE 15.1.4-7

VARIATION OF F/A HOT CHANNEL MAXIMUM CLADDING TEMPERATURE FOR A LOSS OF OFF-SITE ELECTRICAL POWER, SECONDARY SHUTDOWN SYSTEM TRIP

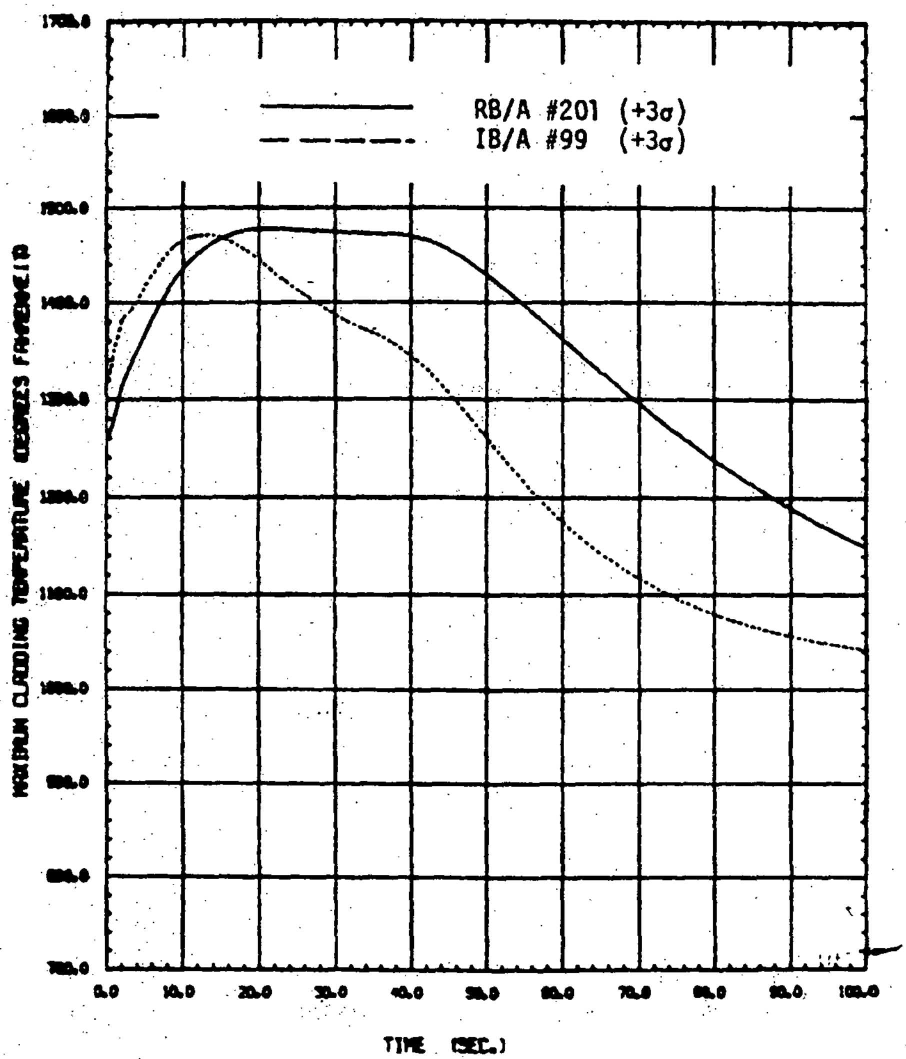

FIGURE 15.1.4-8

VARIATION OF B/A HOT CHANNEL MAXIMUM CLADDING TEMPERATURE FOR A LOSS OF OFF-SITE ELECTRICAL POWER, SECONDARY SHUTDOWN SYSTEM TRIP

15.2 REACTIVITY INSERTION DESIGN EVENTS - INTRODUCTION

In the design approach to safety discussed in Section 15.1.1 it was stated that the design in the second level emphasizes the need to insure and confirm the high reliability of the protection systems and of any component or system whose failure could lead to severe core damage. In keeping with this philosophy this section of the PSAR will examine the response characteristics of the reactor to a series of postulated reactivity insertion events. The reactor response to these events is identified through the resultant hot spot fuel pin cladding temperature. For these accident events either; 1) the resultant cladding temperature will be presented, or 2) it will be shown that the Plant Protection System will limit the reactivity insertion to a value less than a specified enveloping insertion.

51 Based on the discussion presented in Section 15.1.2, the severity of these events can only in-part be discerned by examining the resultant hot spot cladding temperatures. The overall severity of the event, as it effects the cladding integrity, is a function of the sum total of all the accumulated strains imposed on the cladding during its lifetime. Therefore, the severity of any event should be evaluated on a case by case basis using the cumulative damage function. In order to minimize the evaluation process and provide a ready determination of the relative severity of the event, the transients generated in this section can first be compared to the umbrella transient described in Section 4.2.1.3.1.
61 52 51 If the accident transient falls within the time and temperature confines of the umbrella event, the conclusion can be drawn that the design life and safety objectives of the fuel assemblies have been attained. If, however, the resultant clad temperature are beyond the time-temperature confines of the umbrella, then supplementary analysis is required to determine the severity of the event.

The following conservative assumptions and conditions were used for the specific purpose of generating the worst case reactivity insertion transients for this section.

1. All full power cases are for the reactor operating at thermal hydraulic design conditions with a power generation of 975 MWT at 3 loop operation. (Power uncertainties are discussed in Section 4.4.3.2).

2. Since the highest power fuel assembly and smallest Doppler coefficient occur at the beginning-of-equilibrium cycle (BOEC) the transients are analyzed for this particular worst period in core life.

3. With burnup, the power generation and steady state temperature decrease (flows are constant) in the fuel assemblies and consequently, the temperatures due to the transients would decrease.

4. The nominal Doppler coefficient for BOEC is - 0.0062 (see Section 4.3.2.3) however, for overpower transients it is more conservative to take the lower bound value of the ±20% uncertainty on this value. For studies in this section, -0.005 was used for the Doppler coefficient except where noted. (The exceptions being those cases where a larger Doppler yields more conservative results.)

5. Figure 4.2-93 of Section 4.2.3.1.3 of the PSAR includes 0.1 second unlatch time delay between start of CRDM stator current decay and start of the primary control rod motion. For preliminary Plant Protection System transient evaluation, a 0.2 second overall scram delay (see Section 15.2 and 7.2.1.2.3) has been assumed. This scram delay includes PPS logic, scram breaker and the unlatch time delay, leaving sufficient margin
46 on overall PPS response time to assure conservative analysis.

6. The rod worths used to predict post trip negative reactivity insertions are the design expected values for the primary control rods and the minimum expected values for the secondary control
61 rods. (See Section 15.1 for further details.) For both sets of control rods the single most reactive control rod is assumed to be stuck in the withdrawn position. At BOEC the primary control rods negative reactivity insertion capability is less than any later time in the cycle. The purpose of these assumptions is to provide a realistic minimum prediction of shutdown reactivity and hence the slowest rate of power decrease. This provides a conservatively high prediction of reactor temperatures after shutdown.

51 7. Three sigma (3σ) hot channel factors were used for all the analyses and the temperatures shown are the inner surface of the hot pin cladding at the highest temperature position, both axially and circumferentially on the fuel rods. (Position is under the wire wrap).

The possibility of additional fuel-cladding mechanical interaction during rapid reactivity insertion events is acknowledged as indicated in
61 Figure 4.2-22 and subsection 4.2.1.3.1.1. However, present models for transient fuel cladding mechanical interaction are admittedly lacking phenomenologically at prototypic CRBRP design conditions, and therefore, are not used for PSAR analyses. The fuel models in codes used to calculate the effects of core disruptive accidents (e.g. SAS3A), are designed to give initial conditions for calculations of fuel motion from a ruptured rod, rather than to calculate detailed cladding responses to a terminated transient. These codes assume that cladding loads are simple functions fo fuel properties during the transient. Such assumptions are acceptable for determining gross fuel rod behavior during a severe transient (e.g., fail-no-fail), but are insufficient for calculating time varying cladding strain on cumulative damage function during a typical upset event.

As noted in subsection 4.2.1.3.1.3, these models are being developed as part of the LMFBR fuel transient testing program. These models for fuel and cladding behavior during rapid reactivity insertion events, as well as the cladding damage models, will be verified for prototypic CRBRP design and operating conditions. Plans for obtaining test data for formulation and verification of these models are provided in Section 4.2.1.4.1 and 4.2.1.3.1.3 and are referenced in Section 4.2, numbers 53 through 58.

Although only preliminary results have been obtained to date, albeit not completely prototypic, the fact that no significant fuel rod damage has been observed for simulated PPS terminated events (subsection 4.2.1.3.1.3) provides confidence that satisfactory performance can be demonstrated at FSAR submittal. At that time, the variation of fuel operating parameters during normal operation will be used as initial conditions for the rapid reactivity insertion transients.

Since fuel contact pressure was not considered for PSAR analyses, assumptions 2 and 3 remain valid as they would result in the largest temperature increase during these types of transients. It is this temperature swing that is evaluated at the most damaging time in life which, neglecting fuel contact pressure, is at end-of-life when plenum pressure is highest, and fuel rod cladding thickness is at minimum.

The first two of the above restrictions are obvious. The first assures a zero plastic worth if the stress remains below the proportional elastic limit; the second assigns a failure if the ultimate strength is achieved or exceeded.

The third restriction assures that the plastic worth is not counted twice in going from one time interval to the next. For instance, if at the end of the i-th time interval, $\sigma_i \geq \sigma_p$ then a value has been assigned to L_T. Thus if, at the start of the $(i+1)$ interval, σ_i is retained (i.e., $\sigma_{i+1} = \sigma_i$) then no further assignment to L_T is required so long as σ_{i+1} remains constant or decreases (i.e., $\dot{\sigma} \leq 0$) and so long as the proportional elastic limit remains constant or increases (i.e., $\dot{\sigma}_p \geq 0$). Clearly, if, during the $(i+1)$ interval, σ_{i+1} is increasing or if $\dot{\sigma}_p$ is decreasing then an additional assignment must be made to L_T.

Although the conditions for the activation of L_T are perfectly general, it is typically applied only in the case of rapid transient events because at normal steady state conditions the CRBRP fuel-rods operate below the proportional elastic limit. A transient event is operationally defined as an event in which there is a sufficient change in temperature and/or stress but of sufficiently small duration that changes of a metallurgical nature (e.g., compositional changes and additional fluence effects) can be ignored. It follows, therefore, that the only typically encountered source of variation in the proportional elastic limit, as it affects L_T, is sudden change in temperature. 25

The following list of pertinent Thermal-Hydraulic initial conditions were used for the accident events presented in this section:

Primary Flow (LB/sec/loop)	3841
Primary Hot Leg Temperature (°F)	1015*
Primary Cold Leg Temperature (°F)	750*
Intermediate Flow (LB/sec/loop)	3555
Intermediate Hot Leg Temp. (°F)	956*
Intermediate Cold Leg Temp. (°F)	671*

The following is a Summary Table of the events considered in this Section. Table 15.2-1 identifies: 1) the event, 2) the maximum inside diameter cladding temperature resulting from a primary and secondary scram, and 3) comment on the severity of the event.

*These values include an additional 20°F over their normal value to allow for instrument error and control dead band allowance.

TABLE 15.2-1

REACTIVITY INSERTION DESIGN EVENTS

Section No.	Event	Max. Clad. Temp.* Primary Scram	Secondary Scram	Comments
15.2	Reactivity insert design events			
15.2.1	Anticipated Events			
15.2.1.1	Control assembly withdrawal @ startup	NA (See 15.2.1.1)	1383°F	Temp. shown for 1¢/sec. withdrawal. Resultant Temp. less than operating condition. (Full Power)
15.2.1.2	Control assembly withdrawal @ power	1510°F	1610°F	Based on extremely small withdrawal rate - Results are within the guidelines of Table 15.1.2-2.
15.2.1.3	Seismic reactivity insertion (core, radial blanket and control rod) - OBE	1440°F	~1440°F	Based on postulated 30¢ step reactivity insertion - Results are within guidelines of Table 15.1.2-2.
15.2.1.4	Small reactivity insertions	1500°F	1560°F	For 2¢/sec insertion case - Results are within guidelines of Table 15.1.2-2.
15.2.1.5	Inadvertent drop of single control rod at full power	Less than init. cond.	Less than init. cond.	Results fall within guidelines of Table 15.1.2-2.
15.2.2	Unlikely Events			
15.2.2.1	Loss of hydraulic holddown	1415°F	1420°F	Results are within guidelines of Table 15.1.2-2.
15.2.2.2	Core radial movement	1470°F	1510°F	For non-seismic conditions - Results fall within guidelines of Table 15.1.2-2.

*Fuel pin inside diameter cladding temperature (under wire wrap)

TABLE 15.2-1 Continued

	Section No.	Event	Max. Clad. Temp.* Primary Scram	Secondary Scram	Comments
	15.2.2.3	Mal-operation of reactor plant controllers	<1510°F	<1610°F	Less than limiting condition shown in 15.2.1.2-1
	15.2.3	Extremely Unlikely Events			
61	15.2.3.1	Cold sodium insertion	Less than init. cond.	Less than init. cond.	Results fall within the guidelines of Table 15.1.2-2.
61	15.2.3.2	Gas bubble through core	<1480°F	<1480°F	Results fall within the guidelines of Table 15.1.2-2.
61	15.2.3.3	Seismic reactivity insertion (core, radial blanket and control rod) - SSE	<1505°F	NA	Based on postulated 60¢ step reactivity insertion - Results fall within the guidelines of Table 15.1.2-2.
61	15.2.3.4	Control assembly withdrawal at startup-max. mech. speed	NA (See 15.2.3.4)	800°F	For 20¢/sec reactivity insertion - Results fall within the guidelines of Table 15.1.2-2.
61	15.2.3.5	Control assembly withdrawal at power - max. mech. speed	1420°F	1460°F	For 20¢/sec reactivity insertion - Results fall within the guidelines of Table 15.1.2-2.

*Fuel pin inside diameter cladding temperature (under wire wrap)

TABLE 15.2-2

CRITERIA FOR PRELIMINARY EVALUATION OF UPSET AND EMERGENCY EVENTS

Event Classification	Control System	Severity Level	Type of Fault	Type of Assembly	Maximum Temperature (°F)	Reference Figure
Upset (Anticipated Fault)	Primary	Operational Incident	Undercooling	Fuel	1450[1]	4.2-19
				Blanket	1450	4.2-30
			Reactivity Insertion	Fuel	1450[1]	4.2-19
				Blanket	1400	4.2-32
	Secondary	Minor Incident	Undercooling or Reactivity Insertion	Fuel	1600[2]	4.2-20
Emergency (Unlikely Fault)	Primary			Blanket	1600	4.2-33
	Secondary	Major Incident	Use faulted limits of Table 15.1.2-2			

NOTES:

(1) A reevaluation of the referenced fuel rod design transient analysis showed that as many as 3 step changes to 1500°F (start of life) with a 345 second hold time could be accommodated each cycle with no reduction in fuel rod design life.

(2) For temperatures in excess of this value, the transient limit curves of Figures 4.2-21 and 4.2-22 may be used for evaluation.

15.2.1 ANTICIPATED EVENTS

15.2.1.1 Control Assembly Withdrawal at Startup

15.2.1.1.1 Identification of Causes and Accident Description

For this event, it is assumed that the reactor has reached criticality. To reach criticality at any time in the CRBRP core, it is first necessary to completely withdraw the four secondary control rods in Row 4 as well as the two Row 4 primary startup rods. Then to ascend to power requires withdrawal of the remaining primary control rods which are normally sequentially moved to keep them at nearly the same elevation.

It may be postulated that the electronic circuit which produces pulses that control the rod movement malfunctions and that the pulser begins to withdraw one of the primary rods. Although the pulser might malfunction, the reactor operator has visual indications of rod movement and could stop the withdrawal. If the operator fails to stop the movement, an alarm will sound before the rod becomes misaligned over some preset value with the other primary control rods in its bank. The operator must re-align the rods before the alarm is shut-off.

The maximum design rod withdrawal speed is limited to about 9 inches per minute (ipm). If the rod speed exceeds this withdrawal rate, an electronic logic circuit will automatically stop the rod. The occurence of a rod being withdrawn at the maximum design speed is classified as an anticipated fault. A maximum ramp insertion to the core of 2.4 ¢/sec could occur as the rod passes the core midplane (considering the highest worth control rod).

15.2.1.1.2 Analysis of Effects and Consequences

To analyze the effects of a continuous rod withdrawal at startup, the reactor was assumed to be initially operating with a very low power level. For this analysis, 1 Mw, 600°F reactor inlet temperature and 40 percent of full flow where assumed. Beginning of equilibrium cycle core conditions were modeled. The minimum Doppler coefficient of -0.005 was used for the core. This value is obtained by decreasing the nominal Doppler coefficient by 20% for uncertainties as discussed in Section 4.3.2.3. Continuous ramp reactivity insertions of 2, 5 and 20¢/sec were studied.

The secondary control rod system was used for shutdown in the studies (with the maximum worth control assembly assumed to be stuck). Trip was taken at 56 percent full reactor power which adequately accounts for the flux to total flow subsystem performance at this reduced power and flow level. This setting corresponds to a power-to-flow ratio of 1.4. Primary system action would be initiated based on Flux - $\sqrt{\text{Pressure}}$ at approximately the same time. However, since the response of the primary rods is faster, the resulting transient is less severe. [Note: At full power and flow, the secondary control rods would be tripped at a power-to-flow ratio of 1.3.]

The runs were made using FORE-II (see Appendix A). Figure 15.2.1.1-1 shows the variation in reactor power for the various ramp insertion rates. Figure 15.2.1.1-2 thru -4 show the fuel assembly hot pin (with 3σ hot channel factors) maximum fuel, cladding and coolant temperatures. As can be seen, higher cladding temperatures can be experienced for the smaller ramp insertion rates. The reason for this effect is that the slower rate allows more core energy to be developed before the trip signal occurs. The 2¢/sec case resulted in a short duration cladding temperature increase from 600°F to 1307°F which should result in insignificant damage to the pin cladding since its normal full power maximum temperature would be over 1400°F; for 2.4¢/sec the maximum temperature attained would be slightly less than 1307°F as indicated by the trend shown on Figure 15.2.1.1-3. For smaller ramp insertions less than 2¢/sec higher temperatures could be reached (e.g., 1¢/sec gives a 1383°F maximum cladding temperature after 80 seconds); however, these events would be "turned-around" either by the reactor automatic control system or manually by the operator before the temperatures would reach their normal full power values. The latter means of control is viable since it takes a fairly long time for small ramps to increase the reactor power significantly as indicated by the trend shown on Figure 15.2.1.1-1. [Note: The reactor power can be seen to increase less than 1% of full power per second for each ¢/sec of reactivity ramp insertion].

Parametric studies were performed to determine the effect of initial powers of less than 1 Mw and reactor inlet temperatures of less than 600°F. It was found for both of these cases that lower core temperatures would result.

Although, the fuel assembly maximum cladding temperature rise for 2¢/sec was from 600°F to 1307°F, it was found that the hot pin (with 3σ hot channel factors) in the highest power radial blanket assembly would be substantially less. The maximum cladding temperature rise was found to be from 600°F to 992°F. This trend of the transient effect being less severe than that in the fuel assembly would be expected to result for the 5 and 20¢/sec ramp insertions, also.

15.2.1.1.3 Conclusion

As a source for an uncontrolled rod withdrawal incident to occur at startup, one can postulate that the electronic circuit which produces pulses that control the rod movement malfunctions and begins to withdraw one of the primary rods. Furthermore, the reactor operator failed to stop the rod, despite an alarm indicating that one of the rods is mis-aligned with the other rods in its bank. The speed of withdrawal is limited to about 9 ipm by an electronic logic circuit which will automatically stop the rod if this speed is exceeded. Analyses were performed to show the consequences of 9 ipm continuous withdrawal of a maximum worth control rod. This resulted in a maximum reactivity insertion of 2.4¢/sec.

Analyses using the FORE-II computer code show that the maximum fuel pin cladding temperature for the 2.4¢/sec insertion case was 1307°F. Since the normal full power maximum temperature of the cladding at the same position is over 1400°F, the transient should not produce any significant additional degradation of the cladding. A description of how this type event is incorporated in the pin cladding structural design evaluation is given in Section 4.2.1.3.

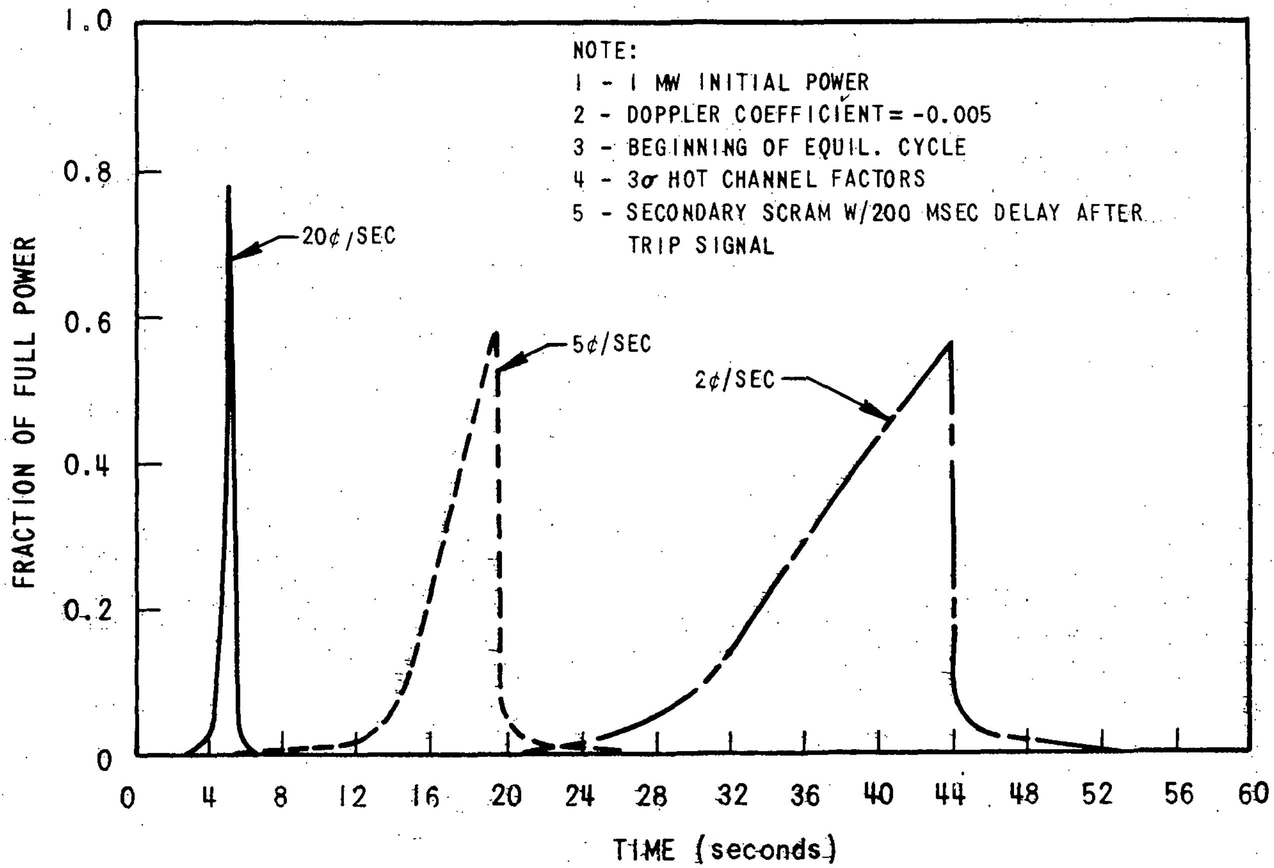

Figure 15.2.1.1-1. Variation in Reactor Power for Control Assembly Withdrawal at Startup

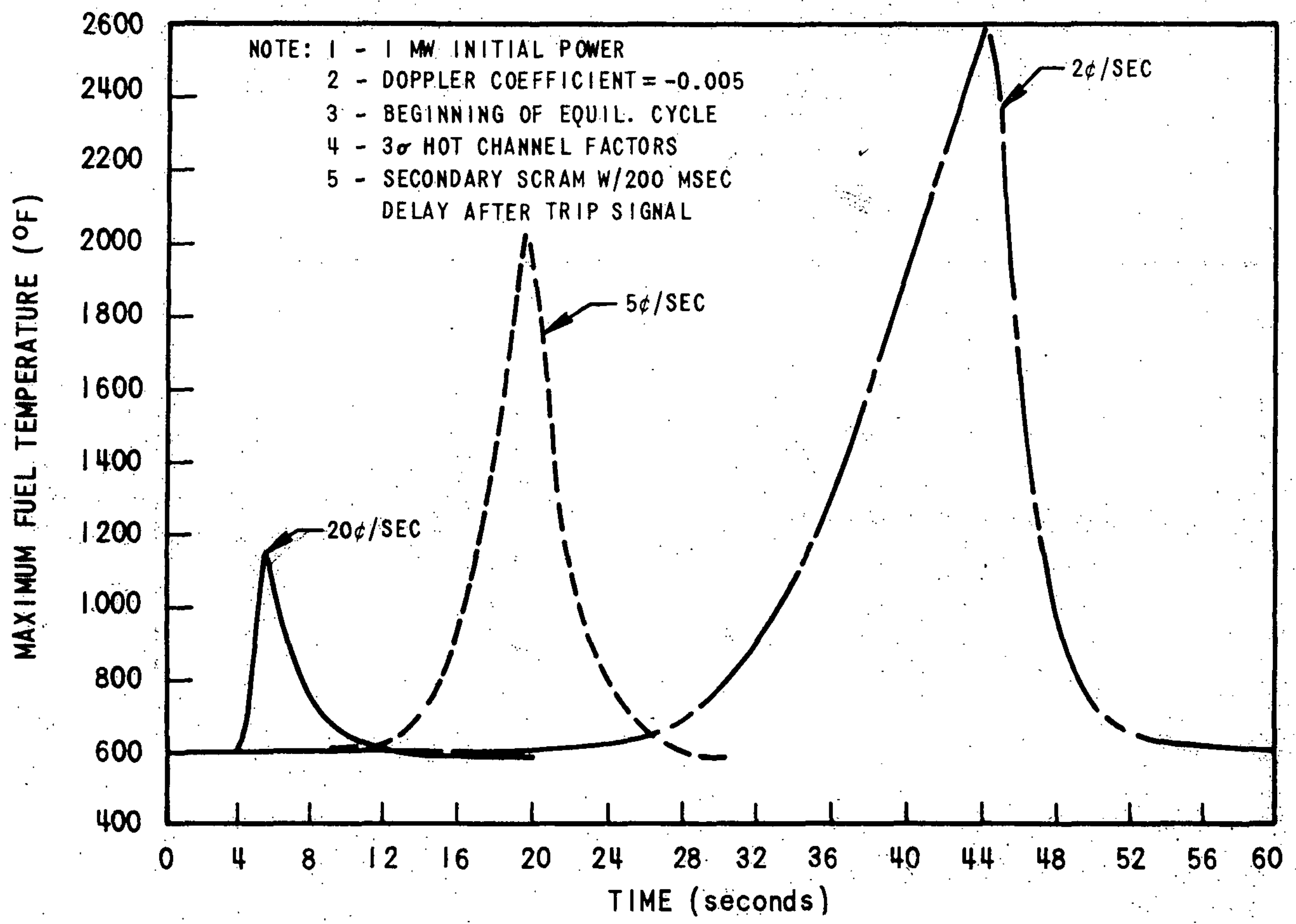

Figure 15.2.1.1-2. Variation in F/A Hot Channel Maximum Fuel Temperature for Control Assembly Withdrawal at Startup

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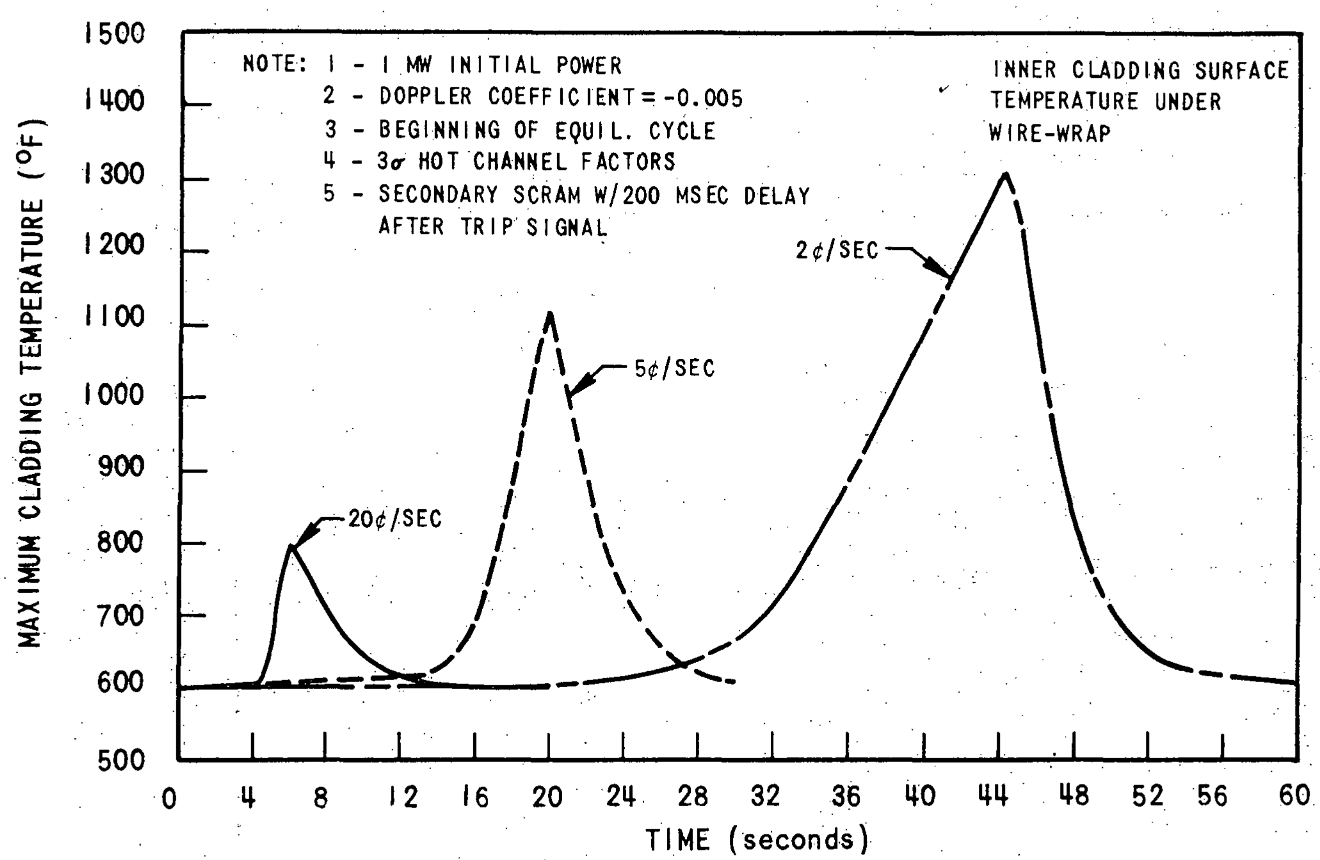

Figure 15.2.1.1-3. Variation in F/A Hot Channel Maximum Cladding Temperature for Control Assembly Withdrawal at Startup

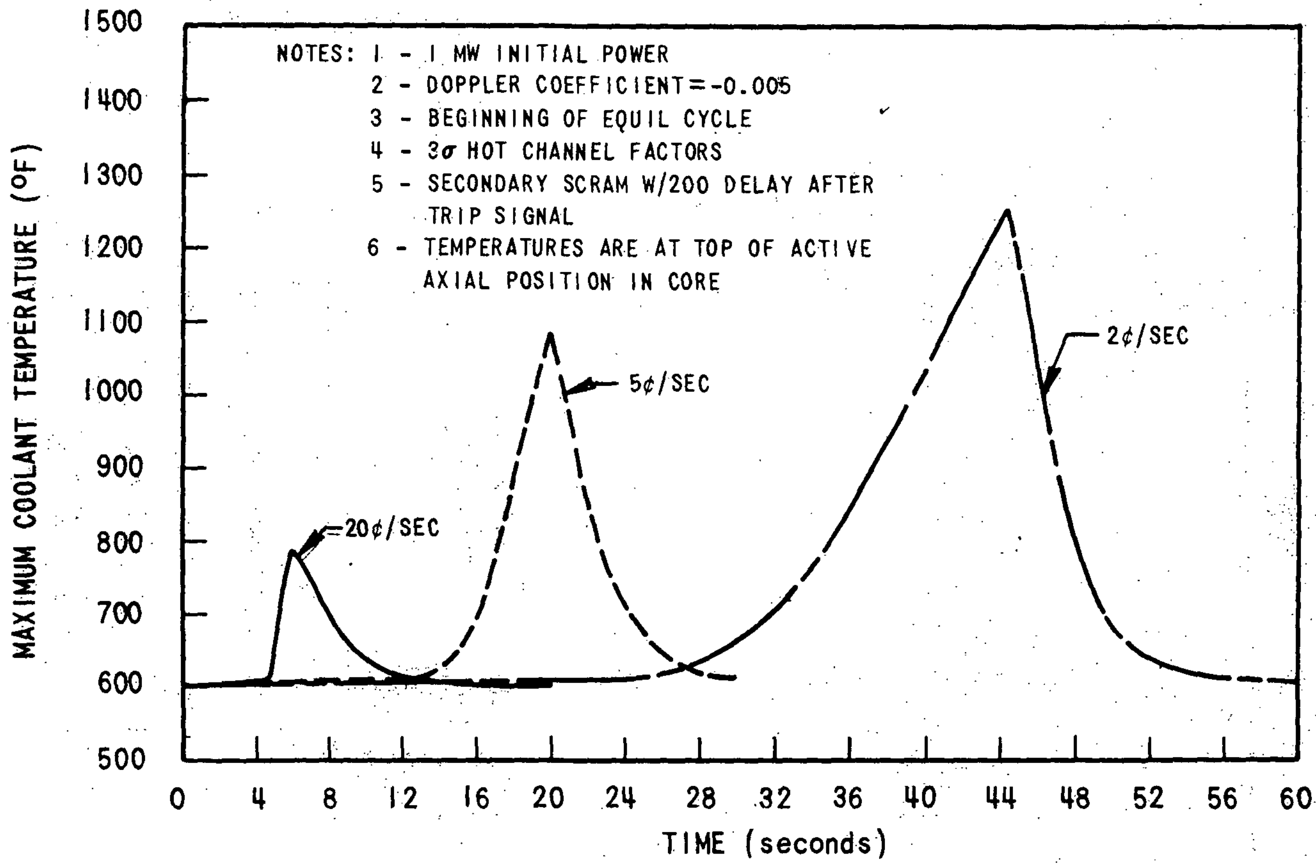

Figure 15.2.1.1-4. Variation in F/A Hot Channel Maximum Coolant Temperatures for Control Assembly Withdrawal at Startup

15.2.1.2 Control Assembly Withdrawal at Power

15.2.1.2.1 Identification of Causes and Accident Description

During full power reactor operation, the in-core primary control rods can be moved by either the automatic control system or manually by the operator. The discussion of rod speed in Section 15.2.1.1 applies here. That is, the maximum design withdrawal speed would be about 9 ipm. The reactivity insertion to the core would have a maximum value of 2.4¢/sec, as also discussed in the earlier section.

The automatic reactor control system and the control rod withdrawal blocks will limit the results of this type event (see Section 7.7). For reactivity insertions of less than approximately 5¢/sec occurring when the controls are in automatic, the automatic control system will correct the results of the event with less than 10% power overshoot and restore the power to the initial condition. At full power rod blocks will limit the maximum power reached for postulated reactivity insertions to a net rate of 5¢/sec or less to under 115% power even for worst case assumptions. For initial powers less than 100%, the power-flow rod block will limit the power excursion to less than 120% of the initial power. These systems are provided to prevent scram for anticipated reactivity faults.

15.2.1.2.2 Analysis of Effects and Consequences

To analyze the effects of a continuous rod withdrawal at power, the reactor was assumed to be initially operating at full power at plant thermal/hydraulic design conditions. Beginning of life and equilibrium core conditions were modelled. The minimum Doppler coefficient (This value is obtained by decreasing the nominal Doppler coefficient by 20% for uncertainties as discussed in Section 4.3.2.3) of -0.005 was used for the core. Continuous ramp reactivity insertions of 2 and 5¢/sec were studied to simulate a range around the postulated insertions. All operator and automatic corrective actions were neglected and reactor shutdown occurred due to scram.

Both the primary and secondary control rod systems were studied separately for their shutdown capability with the maximum worth control assembly assumed to be stuck. For scram with the primary control system, 15% overpower was used for the trip; scram with the secondary system used a trip from a power-to-flow ratio of 1.3.

The runs were made using FORE-II (see Appendix A.) Figures 15.2.1.2-1 through -4 show the variation in reactor power and fuel assembly hot channel (with 3σ hot channel factors) maximum temperatures for the fuel, cladding and coolant. Figure 15.2.1.2-5 shows the maximum cladding temperature for the radial blanket hot pin at the top of the active core axial position in the highest power radial blanket assembly with secondary scram (for a 2¢/sec ramp). As can be seen, the maximum temperature attained is over 100°F cooler than for the fuel assembly hot pin. The second temperature peaks on the figure are due to the slow release of internal stored heat in a radial blanket pin (i.e., radial blanket pins have a 0.52" OD as compared to 0.23" for fuel pins) relative to the flow decay when the primary pumps are tripped. Figure 15.2.1.2-6 summarizes the fuel assembly maximum cladding temperatures for a range of reactivity insertions. As can be seen, higher cladding temperatures can be experienced for the smaller ramp insertion rates. The reason for this effect is that the slower rate allows more core energy to be developed before the trip signal occurs. However, the small ramp insertions (e.g., smaller than 2¢/sec) should be "turned-around" either by the reactor automatic control system or manually by the operator before the temperatures attain values as high as those indicated on Figure 15.2.1.2-6. The latter means of control is viable since it takes a fairly long time for small ramps to increase the reactor power significantly. For instance, a 0.1¢/sec ramp would take about 150 seconds to increase the reactor power by 15%. In any event, the primary rods would scram and prevent maximum cladding temperatures from exceeding 1510°F even for the smallest ramp reactivity insertion (while operating at full power and flow); the corresponding temperature for secondary scram would be 1610°F.

15.2.1.2.3 Conclusions

As a source for an uncontrolled rod withdrawal incident to occur at power, one can postulate that the electronic circuit which produces pulses that control the rod movement malfunctions and begins to withdraw one of the primary rods. However, the reactor operator has visual indications of rod movement and could stop the rod. If the operator failed to stop the rod, the automatic controls or control rod withdrawal blocks will limit the results of the transient such that the reactor power would be restored to its normal level or stopped before a scram signal would be initiated (and less than a 15% power overshoot would result for a full power operation).

Although the uncontrolled rod withdrawal is highly improbable, analyses were performed to show the consequences of a 9 ipm continuous withdrawal of a maximum worth control assembly. This results in maximum reactivity insertions of 2.4¢/sec. FORE-II calculations indicate that the maximum fuel pin cladding temperature for this ramp reactivity insertion would be about 1470°F with primary scram and 1510°F for secondary scram. Since the normal full power maximum temperature at the same position is over 1400°F, the transient should not produce any significant additional degradation of the cladding lifetime capability. A description of how this type event is incorporated in the pin cladding structural design evaluation is given in Section 4.2.1.3.

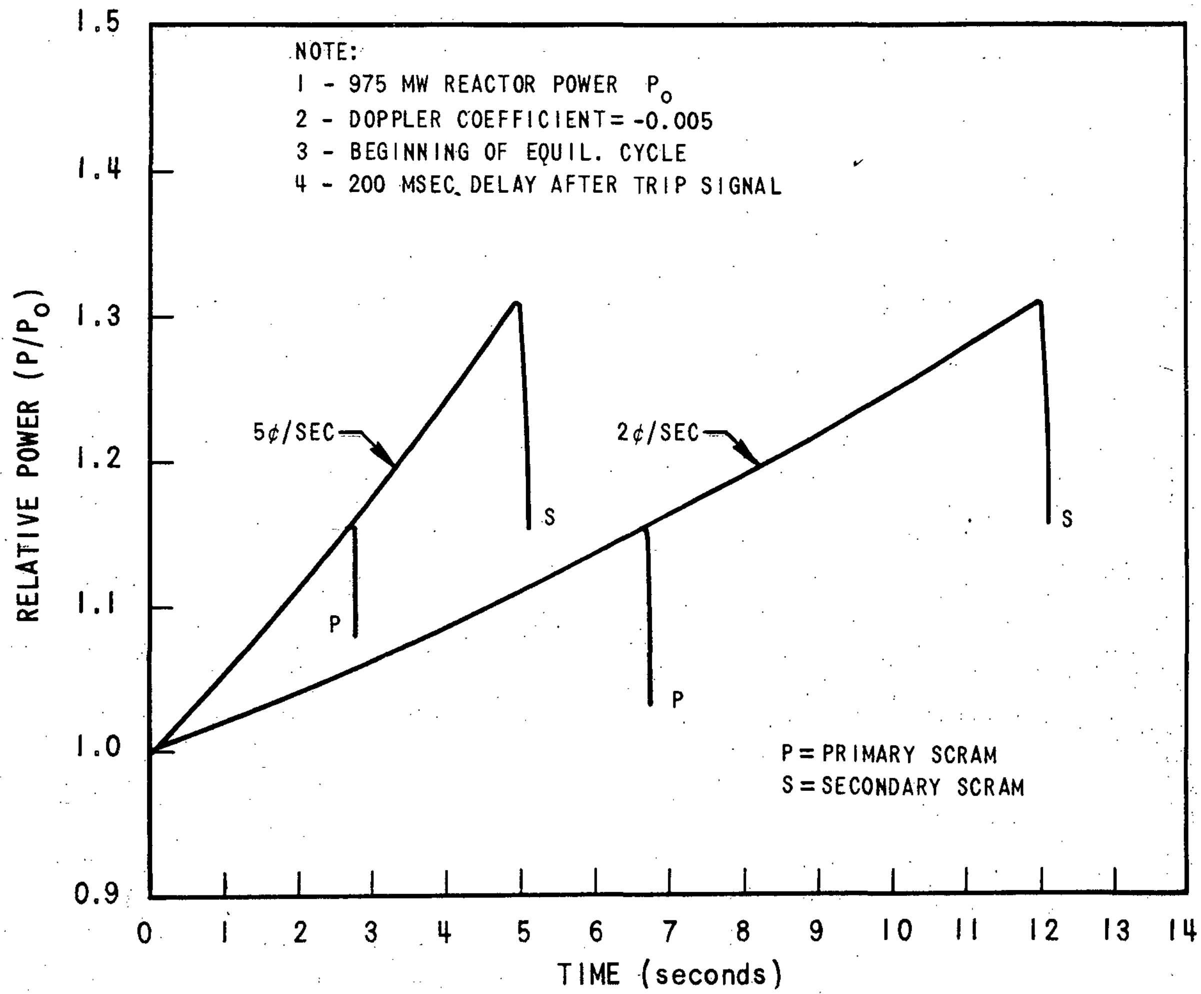

Figure 15.2.1.2-1. Variation in Reactor Power for Control Assembly Withdrawal at Power

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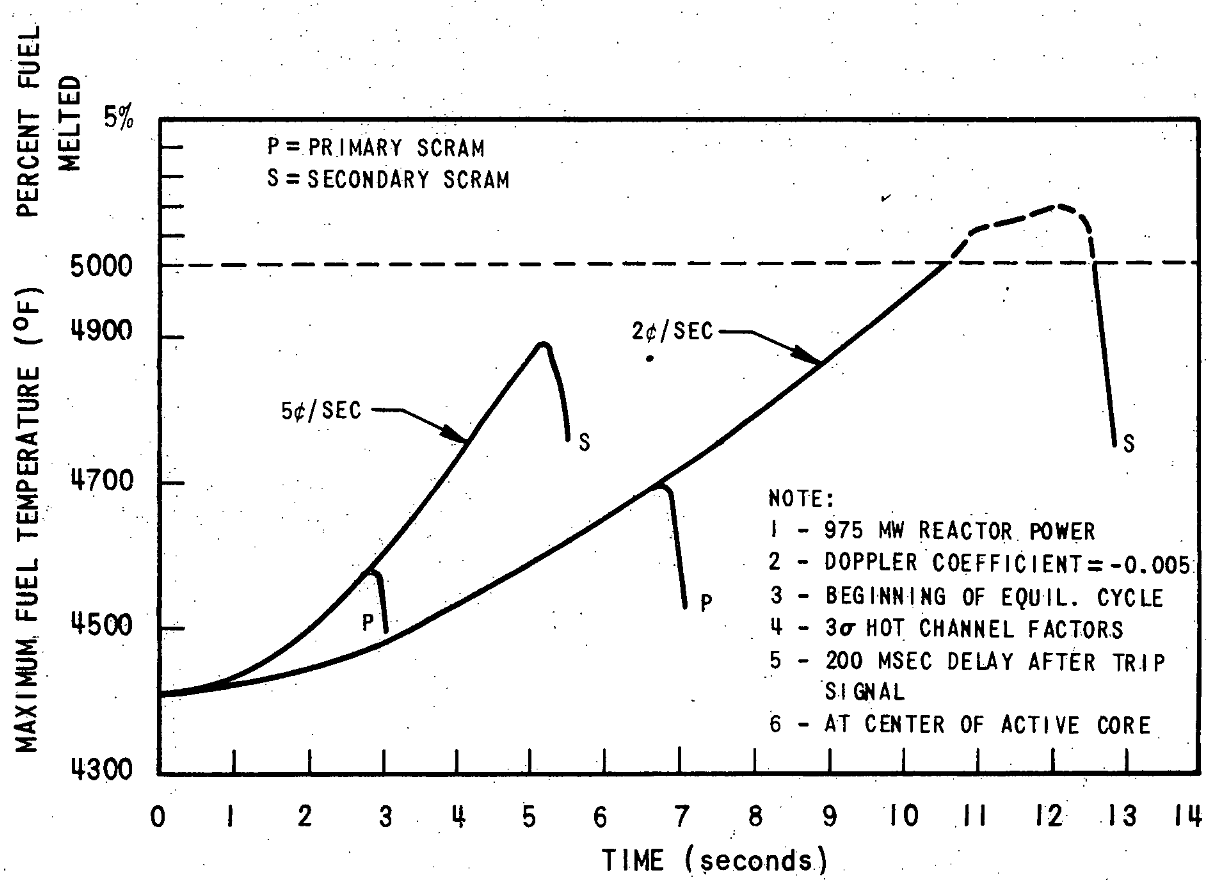

Figure 15.2.1.2-2. Variation in F/A Hot Channel Maximum Fuel Temperature for Control Assembly Withdrawal at Power

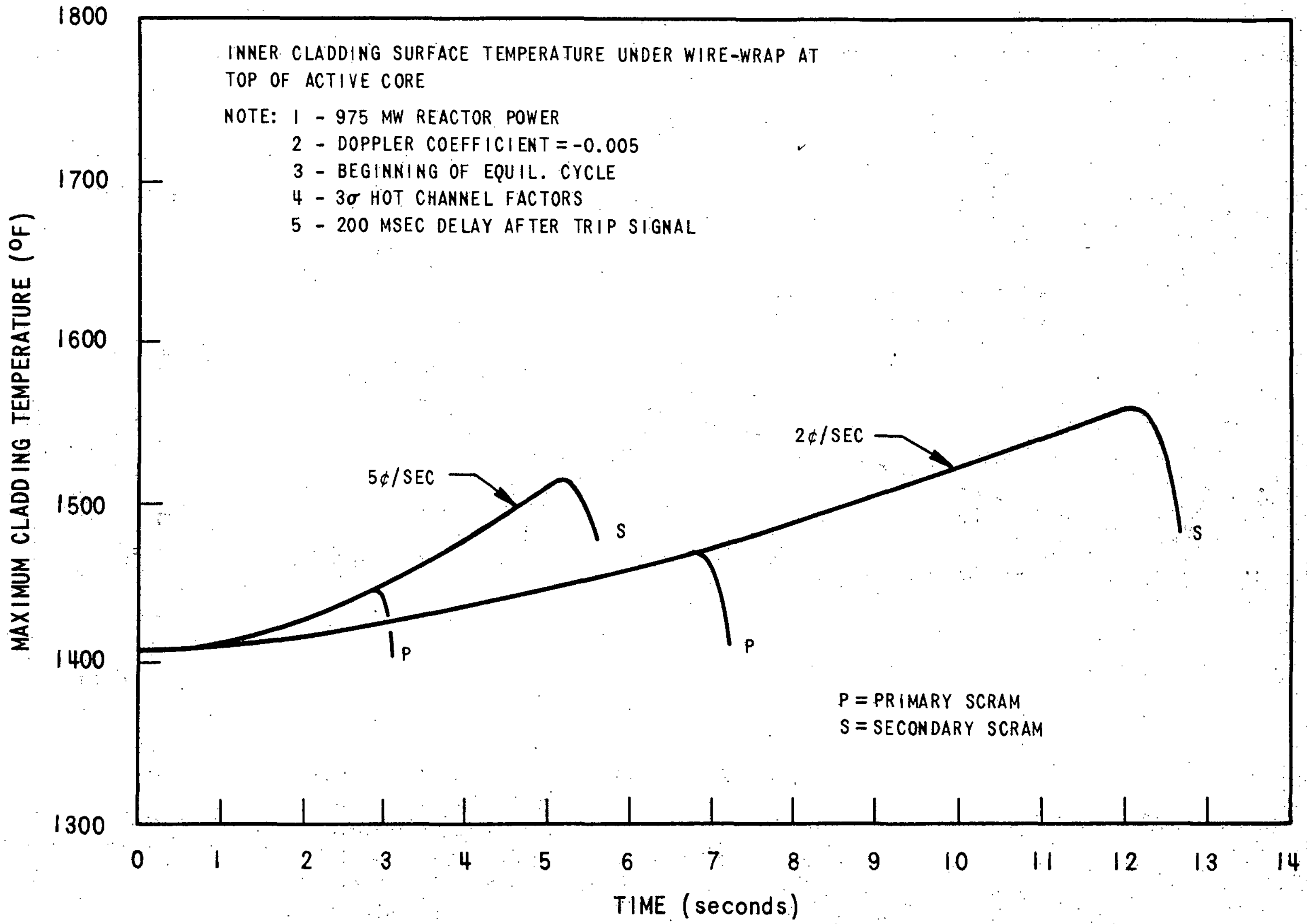

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Figure 15.2.1.2-3. Variation in F/A Hot Channel Maximum Cladding Temperature for Control Assembly Withdrawal at Power

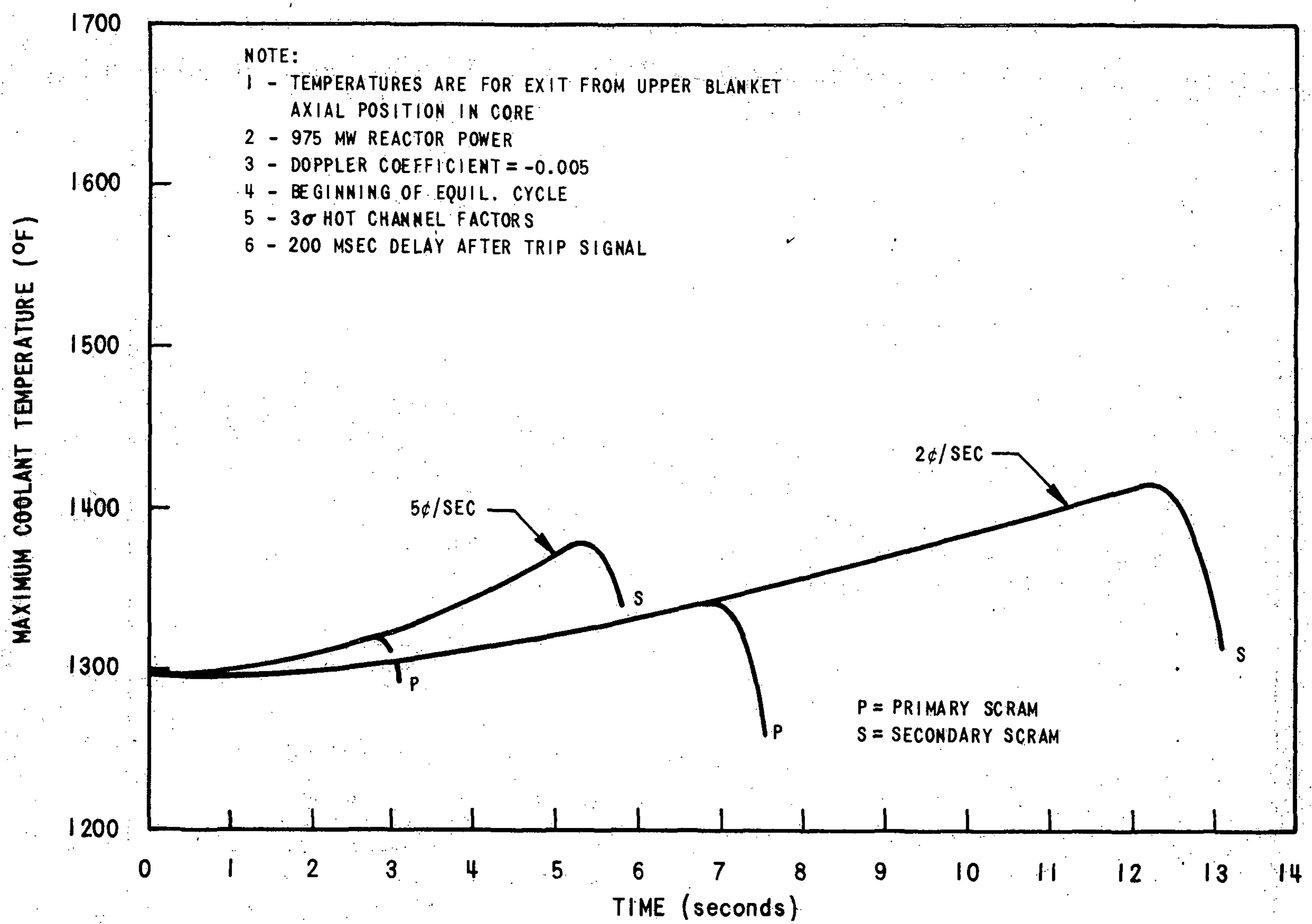

Figure 15.2.1.2-4. Variation in F/A Hot Channel Maximum Coolant Temperature for Control Assembly Withdrawal at Power

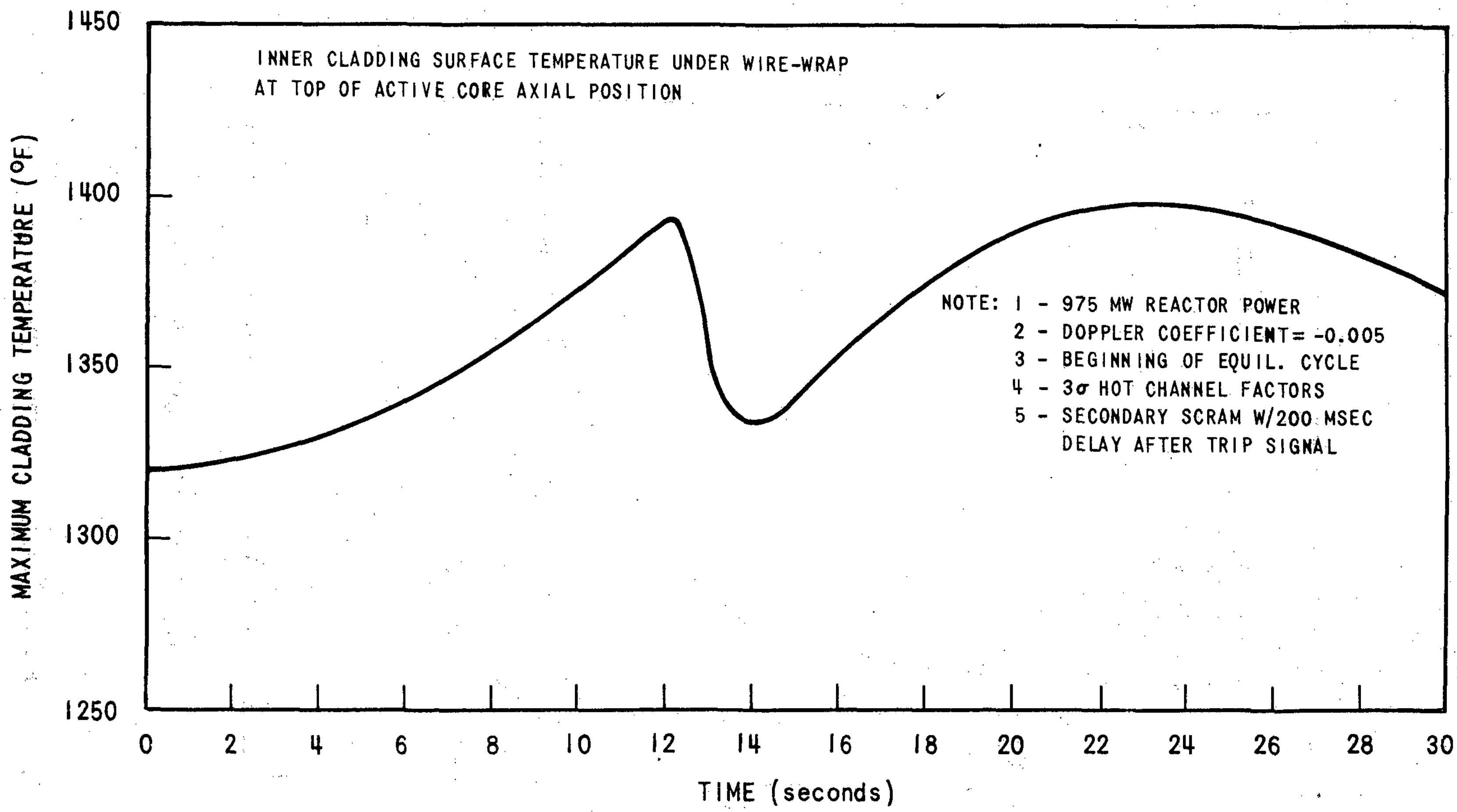

Figure 15.2.1.2-5. Variation in RB/A Hot Channel Maximum Cladding Temperature for $.02/Sec. Ramp Reactivity Insertion

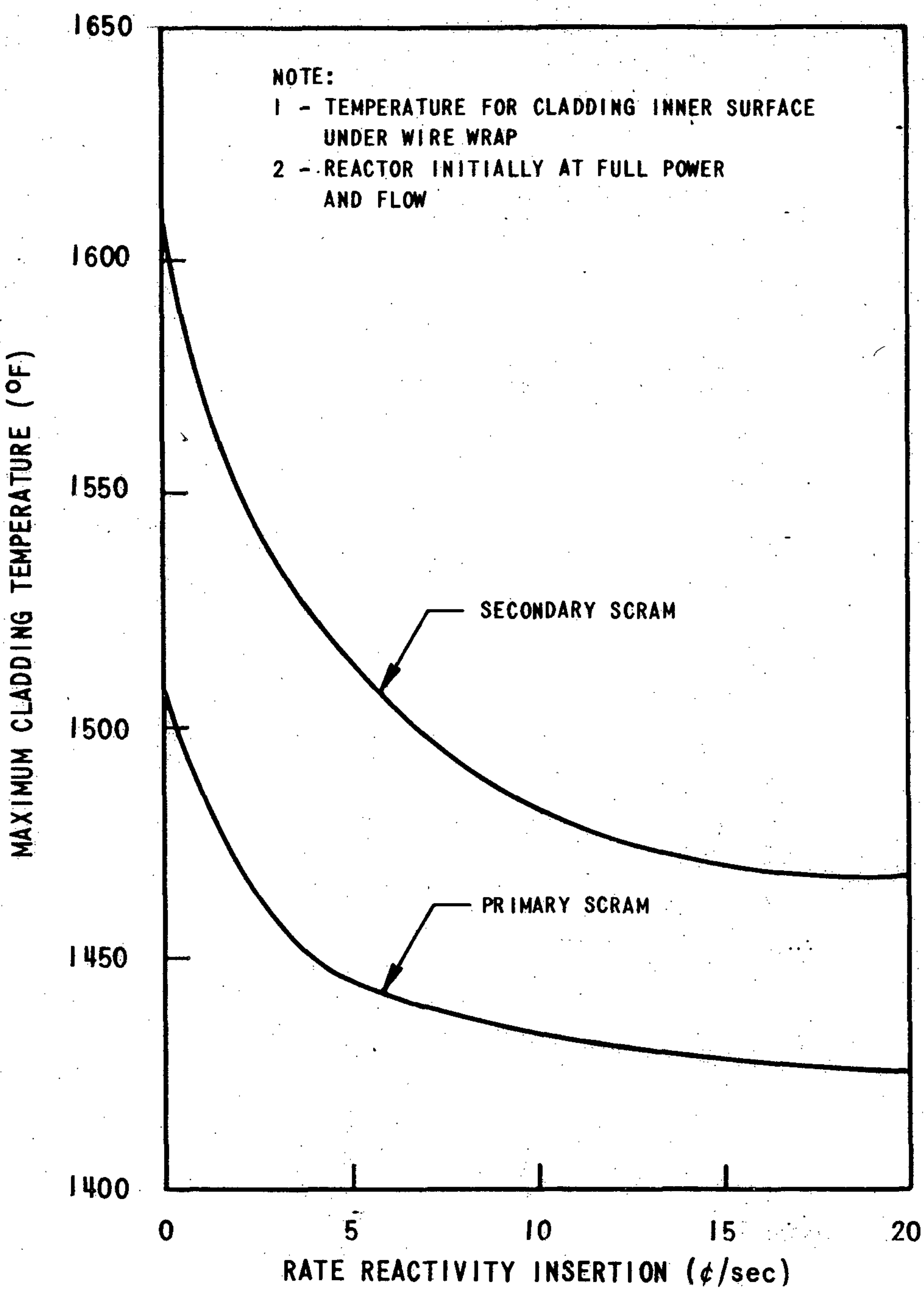

Figure 15.2.1.2-6. F/A Hot Channel Maximum Cladding Temperature Vs. Ramp Reactivity Insertion

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15.2.1.3 Seismic Reactivity Insertion (Fuel, Radial Blanket and Control Rod Assemblies) - OBE

15.2.1.3.1 Identification of Causes and Accident Description

For the Operational Basis Earthquake (OBE) several conditions exist that compound the severity of the event. First, the earthquake could potentially produce a loss of off-site electrical power causing a loss of power to the pumps and consequently a decay of the primary coolant flow. Second, the acceleration forces of the earthquake could potentially cause compaction of the core due to closing of radial gaps between the assemblies at the above core load pad (ACLP) position. This can result in a net positive step reactivity insertion to the core. Third, when the control rods are scrammed, the rate of inward motion is decreased from the normal rate due to a retarding force resulting from seismic induced impacts of the control rod assembly duct and driveline on surrounding guide structures.

There are three sources for producing automatic reactor scram (the operator can also initiate scram). When power to the pumps is lost, a loss of electrical power (LOEP) trip system initiates scram of the primary control rods after a 0.5 second delay. If a step reactivity insertion occurs the primary rods are scrammed when the reactor reaches 115% of full power. A trip signal for the secondary rods would be generated when the power-to-flow ratio reaches 1.30.

The worst combination of the above events with respect to core temperatures during the OBE would be to assume that a step reactivity insertion occurs 0.5 second after the power to the pumps is lost. If the step occurs earlier than this the core flow will be at a higher level when the control rod insertion begins and the resultant temperatures for the event would be lower. This is due to the fact that for steps on the order of 30¢ the power rises very rapidly (in less than 0.1 second) and thus scram would be initiated by either the 15% overpower condition or the 1.30 power-to-flow almost instantaneously. If the step occurs later than 0.5 second the signal due to LOEP would have already started the plant protection system scram and the reactor power would be dropping below its initial value at the time of the reactivity insertion and again less severe temperature conditions would result.

15.2.1.3.2 Analysis of Effects and Consequences

Because the OBE is defined to be of half the intensity of the Safe Shutdown Earthquake (SSE) the more severe insertion rates of the primary control rods for the SSE can conservatively be used for the OBE primary scram analysis. Similarly, other assumptions used for the SSE as will be discussed in Section 15.2.3.3 may be applied to the OBE, and thus the results of the range of step insertions shown in Figure 15.2.3.3-10 may be conservatively used for the OBE. It can be seen that even if a step as large as 30¢ should occur, a fuel assembly hot pin maximum cladding temperature (with 3σ hot channel factors) of less than 1440°F would result. The duration of the cladding temperature above its initial steady state value for this case would be less than 1.0 second. Similarly, a 30¢ step results in a hot pin maximum cladding temperature rise for the highest power radial blanket assembly of less than 25°F.

Insertion rate data for the secondary control rods is not currently available for either the SSE or OBE. Results for the OBE were thus not calculated at this time. If, however, as was the case for the primary control rods, the insertion speeds for the secondary system under normal and seismic conditions are not much different, then the expected maximum cladding temperatures for the OBE will be about the same with either primary or secondary scram. This trend can be seen by comparing primary and secondary scram results for step insertions described in Section 15.2.2.1 (i.e., for a 30¢ step the difference in the maximum cladding temperatures for the two scrams is less than 10°F).

15.2.1.3.3 Conclusions

A conservative analysis was discussed for the anticipated event of an OBE considering the compound effects of core flow decay due to loss of power to the pumps, step reactivity insertion due to changes in core configuration and the decrease in the control rod insertion rate.

It was found that, even if a 30¢ step reactivity insertion occurred for the OBE, the fuel assembly maximum cladding temperature would not exceed 1440°F for the primary scram. Since the normal full power maximum temperature of the cladding is over 1400°F, the transient should produce no significant additional degradation of the cladding lifetime capability. A description of how this type event is incorporated in the pin cladding structural design evaluation is given in Section 4.2.1.3.

15.2.1.4 Small Reactivity Insertions

15.2.1.4.1 Identification of Causes and Accident Description

The only identified source for inserting a significant amount of reactivity in a continuous manner is by withdrawing a control rod. This transient is discussed in detail for occurrence at startup and full power in Sections 15.2.1.1, 15.2.1.2, 15.2.3.4 and 15.2.3.5 and results for ramp insertions from very small rates to 20¢/sec are presented. Small step reactivity insertions can be postulated to occur due to displacements of assemblies that change the core configuration. If the displacements occur quickly and result in a more reactive core configuration, they can be conservatively assumed to produce a positive step reactivity insertion to the core. Sizes of these displacements and reactivity insertions and the effects on the core are further discussed in Sections 15.2.2.1: Loss of Hydraulic Holddown and 15.2.2.2: Core Radial Movement.

15.2.1.4.2 Analysis of Effects and Consequences

An analysis was performed to show the effects of small ramp reactivity insertions (10¢/sec or less) and small step reactivity insertions (10¢ or less). Since the highest power fuel assembly occurs at the beginning-of-equilibrium cycle (BOEC), the transients were analyzed for this particular worst period in core life. By taking the highest power subassembly for BOEC, the analysis is being performed for the highest power condition of all core cycles, which gives a maximum temperature rise (based on power comparisons and considering same gap conductance for all rods) in the fuel and cladding, due to a given power increase. By using a BOL gap conductance with a 3σ reduction due to uncertainties, the minimum gap conductance throughout life is being considered (see Section 4.4.2.6.5). The transient temperature changes (ΔT's) in the fuel, cladding and coolant calculated with these highest power/minimum gap conductance conditions are used as a reasonable representation of transients in structural evaluations of the cladding throughout life. That is, the calculated ΔT's are considered to occur at the most damaging time in life which, neglecting fuel contact pressure, is at the end-of-life when the rod's plenum pressure is highest and the cladding effective thickness is at a minimum. If ΔT's for end-of-life power generation had been used, the fuel temperature increase would be less due to the power decrease with burnup; the cladding temperature increase would be less (assuming a BOL minimum gap conductance) due to the thinner cladding (with less temperature gradient through its thickness) and the lower power generation.

The transient change in fuel temperatures is the worst case for the highest power condition with minimum gap conductance. However, the cladding temperature is increased slightly if a higher gap conductance is used in conjunction with BOEC power generation. To demonstrate the magnitude of the effect, a worst case overpower transient (Extremely Unlikely Event, 60¢ step reactivity insertion occuring under SSE conditions, Section

15.2.3.3) was analyzed both with the minimum gap conductance and with twice this value. With the larger gap conductance, the maximum fuel temperature was reduced 300°F, and the maximum cladding temperature was increased 16°F, relative to those for the gap conductance which was used in the Section 15.2 analysis. Although the cladding temperature rise is slightly greater for gap conductances in excess of those used in Section 15.2, the substantially greater increase in fuel temperature is felt to be more important, particularly for evaluations of fuel-cladding contact pressures during overpower events.(*) In addition, it is important to calculate worst case maximum fuel temperatures during transients to evaluate the potential of having molten fuel generated in the hot rods.

A minimum value Doppler coefficient of -0.005 was used for this core condition. For cases that would cause reactor scram, studies were performed for both primary and secondary control rod shutdown. Scram with the primary control rods was taken at 15% overpower and at a power-to-flow ratio of 1.30 for the secondary control rods. For both systems, the maximum worth control assembly was assumed to be stuck and the shutdown worth was decreased by the appropriate amount.

The calculations were made with FORE-II (see Appendix A). Figures 15.2.1.4-1 through 15.2.1.4-3 show the variation in reactor power, maximum cladding temperature and maximum fuel temperature for 2, 5 and 10¢/sec ramp reactivity insertions. As can be seen, both primary and secondary scram results are shown. Figures 15.2.1.4-4 through 15.2.1.4-6 show the variation in reactor power, maximum cladding temperature and maximum fuel temperature for 5 and 10¢ steps. The highest clad temperatures on Figures 15.2.1.4-4 through 15.2.1.4-6 occur for the 2¢/sec case where a maximum inner surface cladding temperature of about 1560°F can be attained if secondary scram is considered (primary scram would give a maximum cladding temperature of about 1500°F). The trend shown by Figure 15.2.1.4-2 is that higher cladding temperatures result for smaller ramp reactivity insertions. The reason for this effect is that the smaller insertion rates allow more core energy to be developed before the trip signal occurs. However, the smaller insertion cases should be "turned-around" either by the reactor automatic control system or manually by the operator before temperatures can attain values as high as those shown by Figure 15.2.1.4-2. This latter means of control is viable since it takes a fairly long time for small insertions to increase the power

(*) Note that Figure 4.4-33 shows an inner cladding surface temperature decrease in excess of 50°F due to local power depletion of about 13% during a single equilibrium cycle of the hot rod. This would decrease the cladding ΔT in the aforementioned SSE transient by 16°F (for the same cladding thickness and minimum gap conductance) which is about the same as the temperature increase due to the factor of two increase in gap conductance. Assuming a factor of four increase in gap conductance and a 13% power decrease would indicate only a 12°F increase in the Section 15.2 temperatures. Thus, by assuming BOEC power generation/minimum gap conductance conditions, the calculated temperatures are felt to be conservative estimates when used in events which consider cladding temperature alone (worst case at end-of-life as indicated earlier). 25

significantly. For instance, a 0.1¢/sec ramp would take about 150 seconds to increase the reactor power by 15%. Figure 15.2.1.4-4 shows that for even a 10¢ reactivity step it takes over 40 seconds for the reactor power to increase 15%; for the 5¢ reactivity step the power would not increase over 8%. [Note: Since it was desired only to parametrically show the effects of small step reactivity insertions, scram was not considered here. However, the reactor would be shutdown by primary scram when the power reached 115% full power for the 10¢ step case as indicated on Figure 15.2.1.4-4.] As with the ramp insertion cases, either the reactor automatic control system or the operator would bring the reactor back to its initial 100% power level before the temperatures reached values as high as those shown on Figure 15.2.1.4-5.

Several worst cases were also run for the hot pin in the highest power radial blanket assembly. For the 2¢/sec ramp reactivity insertion with secondary scram, a maximum cladding temperature of 1410°F was found. A step insertion of 10¢ was found to produce a 1437°F maximum cladding temperature.

15.2.1.4.3 <u>Conclusions</u>

The consequences of various small size ramp and step reactivity insertions are presented. Analyses are given which conservatively neglect the attenuating effects of the reactor automatic control system or operator corrective action.

For a 2¢/sec ramp reactivity insertion, the fuel assembly hot pin maximum cladding temperatures of about 1500°F and 1560°F were found for primary and secondary scram, respectively. For a 10¢ step insertion, a maximum cladding temperature of less than 1525°F would occur (neglecting the primary trip). Radial blanket hot pin maximum cladding temperatures were found to be significantly cooler for these same transients. A description of how this type event is incorporated in the pin cladding structural design evaluation is given in Section 4.2.1.3.

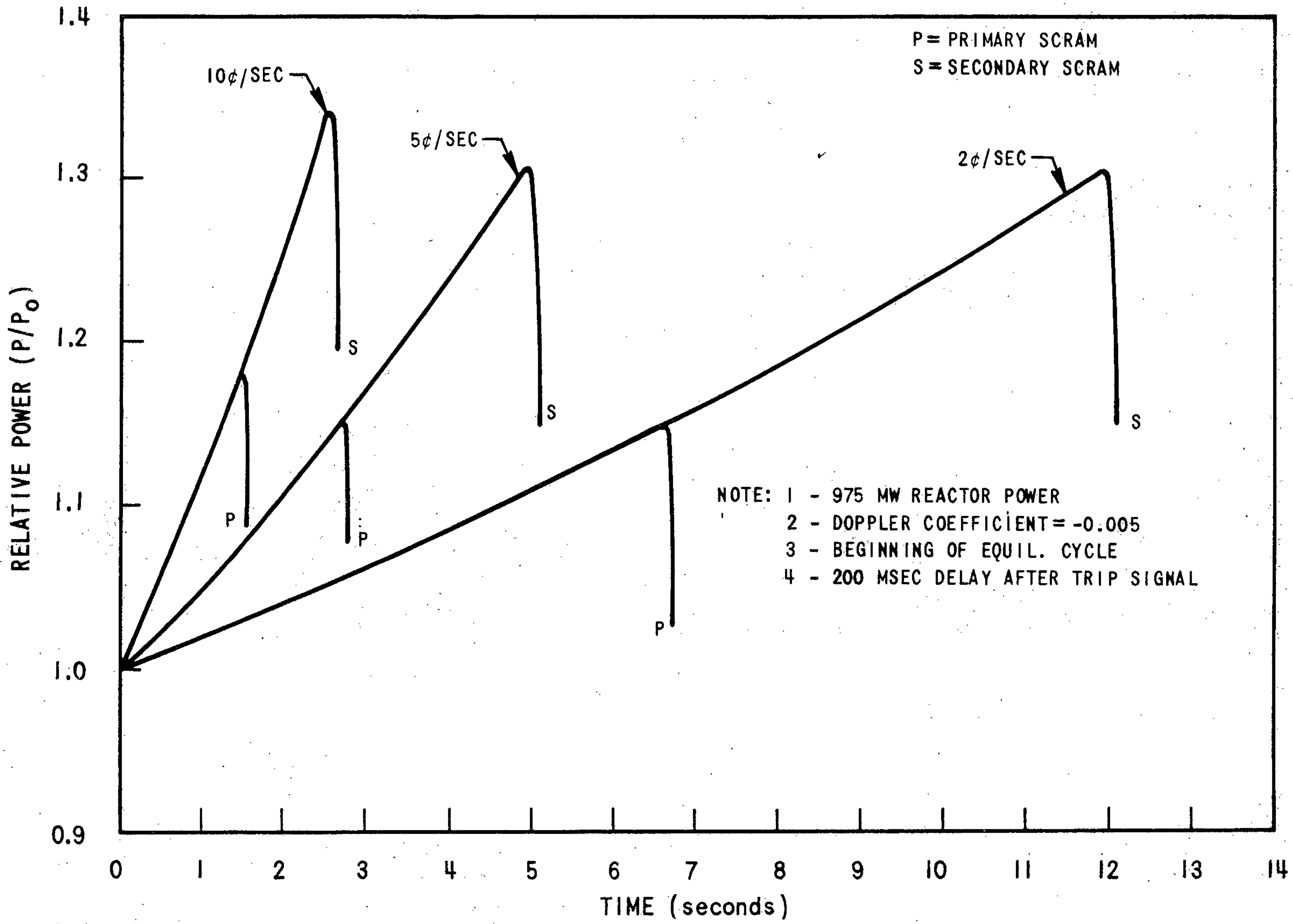

Figure 15.2.1.4-1. Variation in Reactor Power for Small Ramp Reactivity Insertions

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6727-20

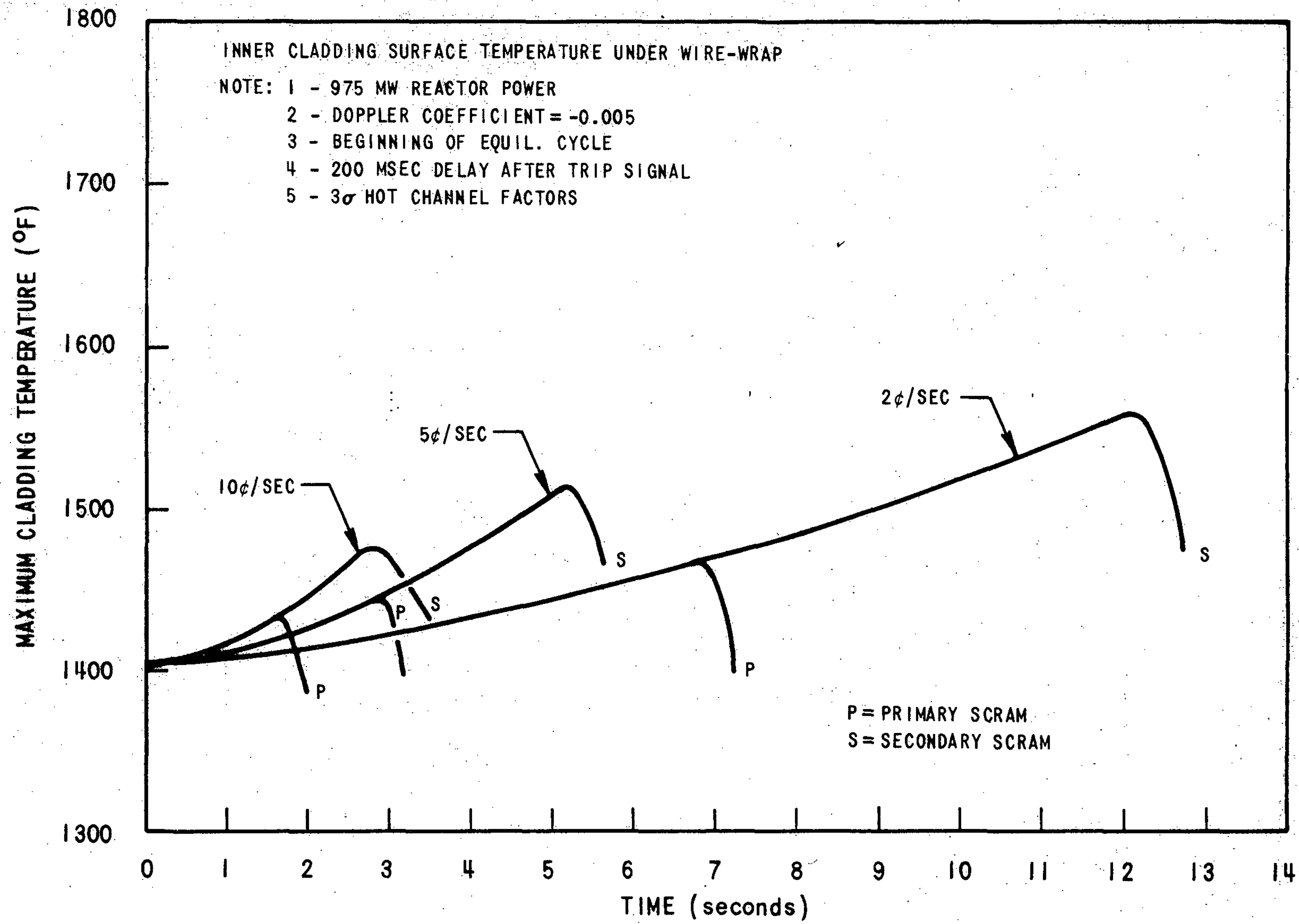

Figure 15.2.1.4-2. Variations in F/A Hot Channel Maximum Cladding Temperature for Small Ramp Reactivity Insertions

6727-63

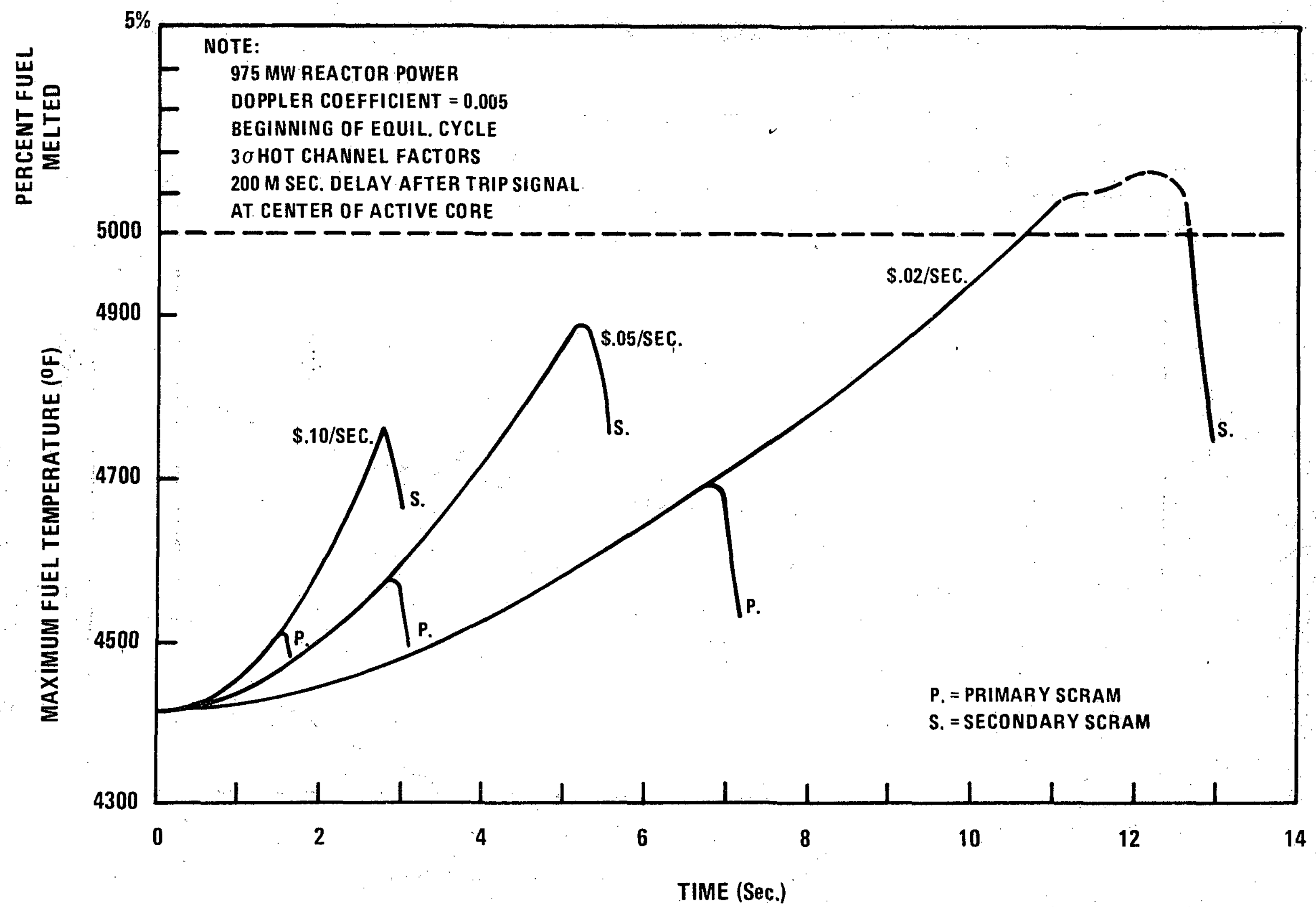

Figure 15.2.1.4-3. Variation in F/A Hot Channel Maximum Fuel Temperature for Small Ramp Reactivity Insertions

6727-21

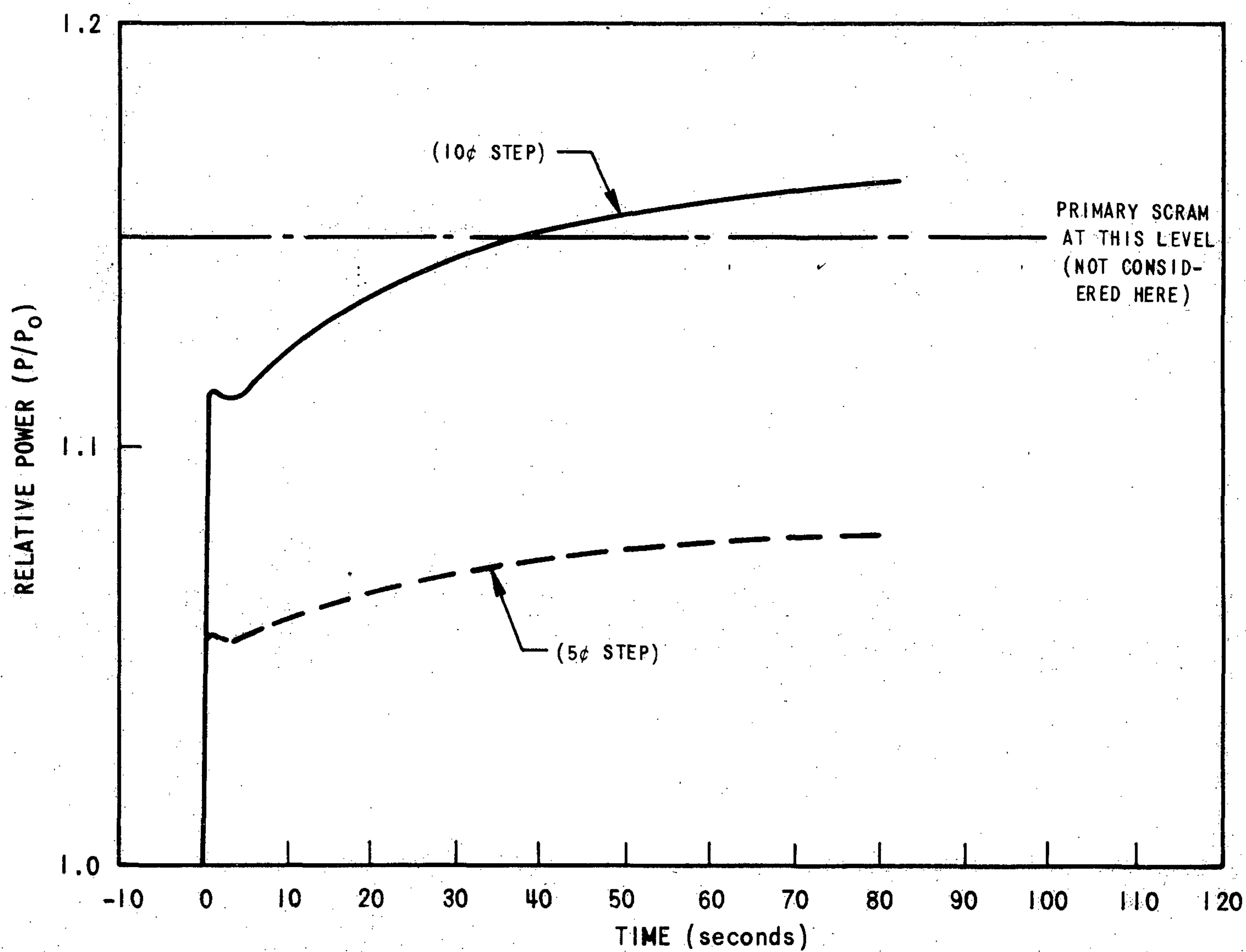

Figure 15.2.1.4-4. Variation of Reactor Power for Small Step Reactivity Insertion

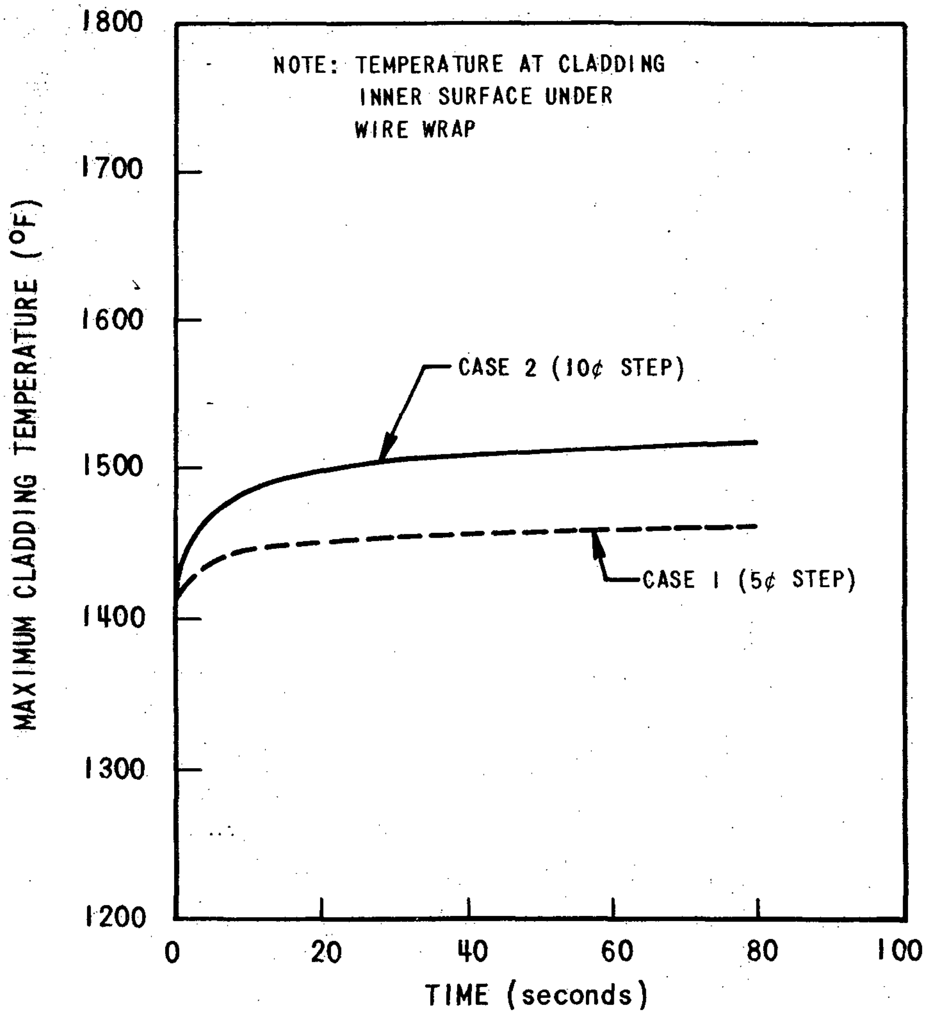

Figure 15.2.1.4-5. Variation of Maximum Cladding Temperature for Small Step Reactivity Insertion

6727-22

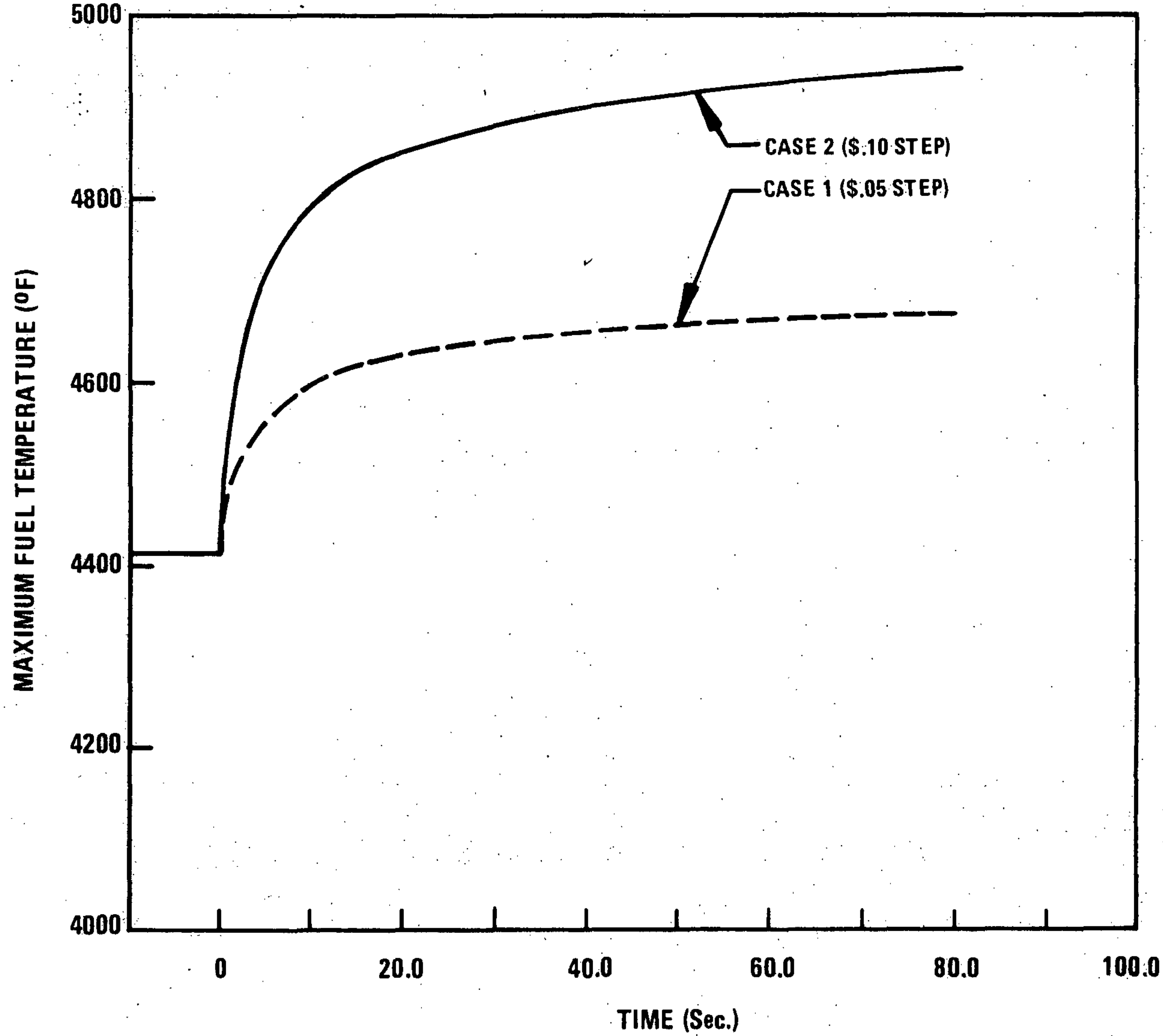

Figure 15.2.1.4-6. Variation of Maximum Fuel Temperature for Small Step Reactivity Insertions

15.2.1.5 Inadvertent Drop of a Single Control Rod at Full Power

15.2.1.5.1 Identification of Causes and Accident Description

The inadvertent drop of a single control rod is postulated to occur as a result of an electrical fault in the affected control rod drive mechanism (CRDM) or its power supply connecting wiring that causes a loss of holding current to the mechanism, or a mechanical failure in the CRDM causing the control rod to be released. The result is a rapid decrease in reactor nuclear power and core temperatures.

To assure a conservative evaluation of the postulated event, the following conditions were applied:

a. Minimum decay heat for a beginning of life core with low prior power history.

b. Maximum worth for the dropped rod corresponding to the equilibrium core center rod. One σ uncertainty with respect to insertion was included in the center rod worth resulting in $3.25 for the dropped rod.

c. Slowest flow coastdown rate after pump trip corresponding to the largest sodium coolant pump rotating kinetic energy and the lowest reactor pressure drop in the design specifications.

These conditions produce the most rapid down transient in reactor temperatures.

The actions of the primary and secondary shutdown systems are as follows:

a. Primary trip - Flux-to-delayed-flux trip occurring at 0.21 second after the malfunction. This delay time includes time for nuclear flux and flux rate to reach the trip point, lags in its measurement, and lags in the trip transfer function.

b. Secondary trip - Trip at 50% nuclear flux occurring at 0.56 second after the malfunction. Delay time includes appropriate delays and lags as above. It should be noted that this preliminary trip function provides a conservative envelope for the performance of the modified nuclear rate subsystem described in Section 7.2.1.2.2

15.2.1.5.2 Analysis of Effects and Consequences

Results of the rod drop analysis, which was generated with the Demo Code, are presented in Figures 15.2.1.5-1 and 15.2.1.5-2. As shown the dropped rod produces a power reduction and consequent drop in Reactor Temperature. Because of the quick action of the PPS, the resulting temperature transient is similar to a normal reactor trip. In the event that the Primary Shutdown System should fail to operate Figure 15.2.1.5-2 shows the Secondary Shutdown System trips the plant with results similar to those for a primary shutdown system trip.

The rod drop study also investigated the drop of control rods of less worth. All resulting transients were less severe than the maximum-worth case shown in Figures 15.2.1.5-1 and 15.2.1.5-2. This remained true even for cases in which the primary trip was neglected and the worth was made small enough to avoid actuation of the secondary trip, resulting in a no-trip event.

15.2.1.5.3 Conclusions

From the results of the analysis presented in Figures 15.2.1.5-1 and 15.2.1.5-2, it is concluded that the single rod drop transient is not significantly more severe than the conventional plant trip shutdown transient.

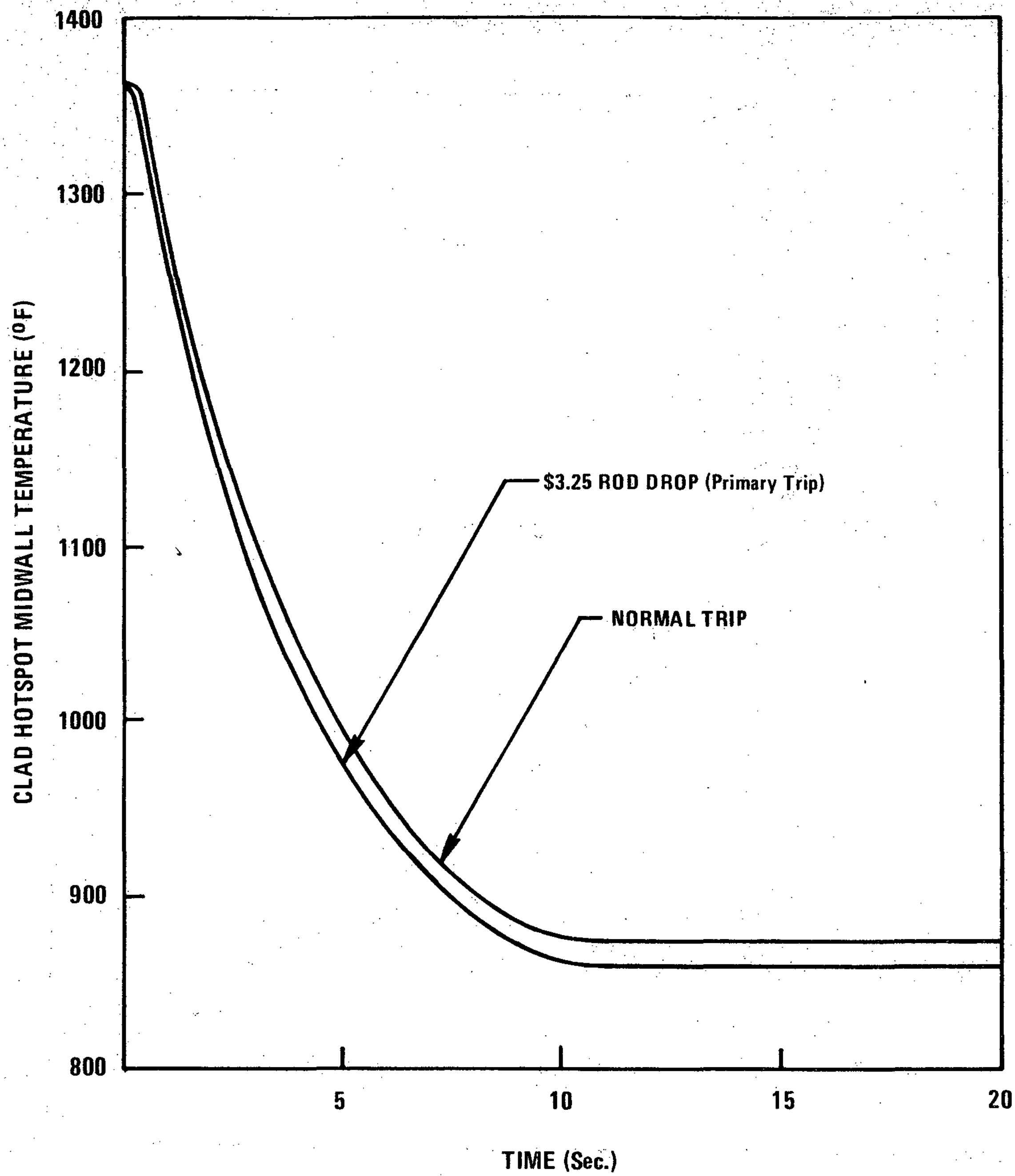

Figure 15.2.1.5-1. Hotspot Clad Midwall Temperature as a Function of Time, Comparison of Rod Drop and Normal Trip

6727-65

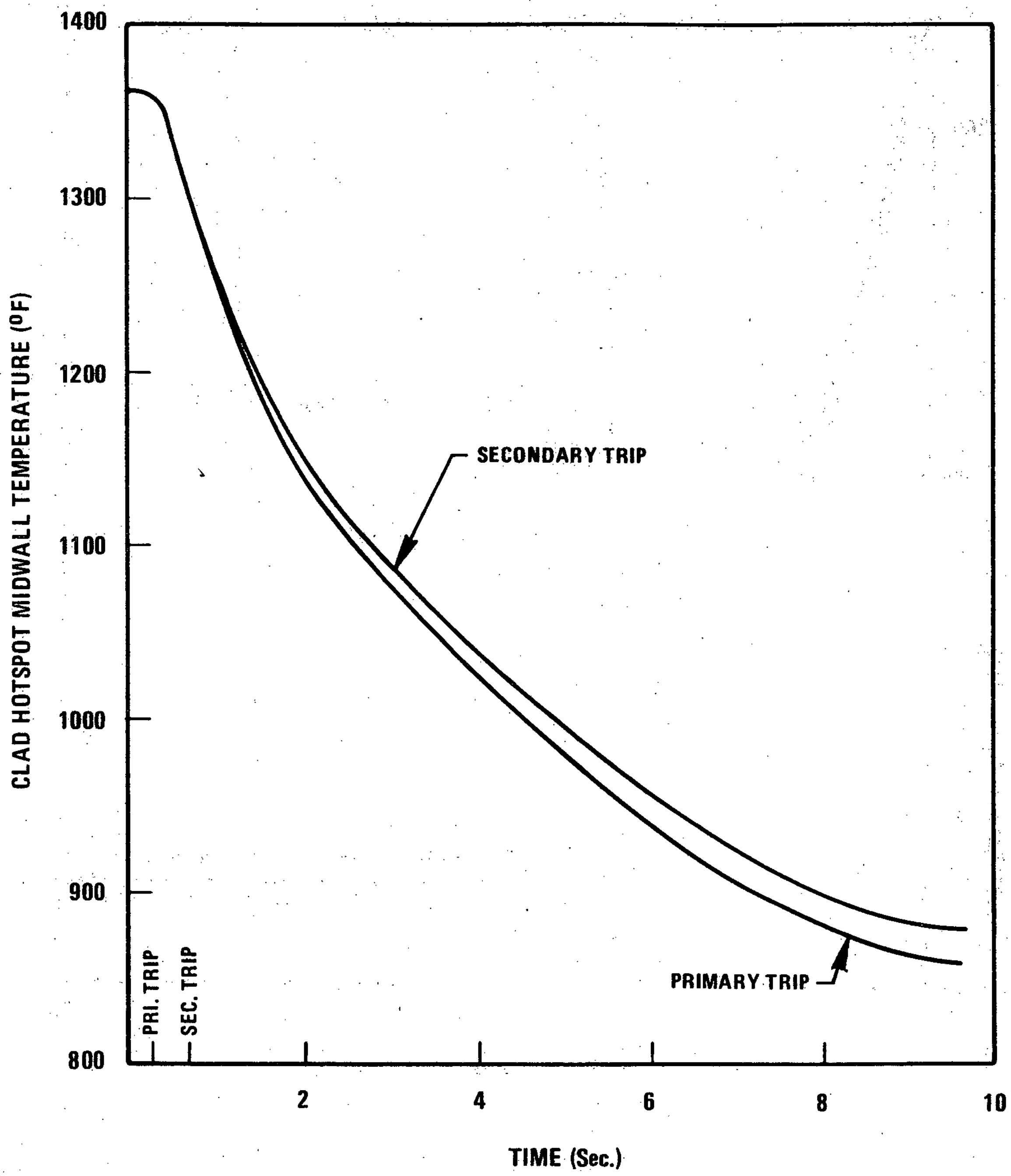

Figure 15.2.1.5-2. Hotspot Clad Midwall Temperature as a Function of Time After a Rod Drop

6727-66

15.2.2 Unlikely Events

15.2.2.1 Loss of Hydraulic Holddown

15.2.2.1.1 Identification of Causes and Accident Description

The function of the hydraulic balance system is to equalize (balance) the pressure forces on the fuel/blanket/control assemblies so that there is little or no lifting force and that the net resultant force (including assembly weight) is downward and approximately equal to the resultant buoyant weight of the assembly (see Section 4.4.2.5 for details). This is accomplished, as shown in Figure 15.2.2.1-1, by exposing a significant portion of the bottom of the assembly to outlet pressure whereas the inside bottom of the assembly is exposed to reactor inlet pressure. A seal is provided between the assembly nozzle and the core support structure receptacle by a piston ring located above and small radial clearance below the entrance slots to the assembly. This seal separates inlet from outlet pressures. Several conditions which could result in the loss of hydraulic balance and cause the fuel/blanket/control assemblies to float upward are considered. They include: inadvertent pump startup during refueling operation, pressure surge associated with a check valve slam, and blockage of the conduit (3/4 inch duct - Figure 15.2.2.1-1) providing communication between the reactor high and low pressure regions.

Loss of hydraulic holddown for either a primary or secondary control rod under operating conditions would have negligible adverse effects since the absorber section is rigidly held in place by a driveline. Consequently, the loss of hydraulic holddown of the control assemblies is not given further attention for at power conditions.

Inadvertent Pump Startup During Refueling

Starting the CRBRP primary pumps requires five steps carried out in proper sequence. First, all scram signals must be cleared. Specifically, the vessel level sensors must be reconnected which implies that the reactor head rotating plugs are in the operating configuration. Second, the primary scram breakers and secondary logic drivers must be manually reset at local stations. Third, all pump auxiliaries (lube oil system) must be started. Fourth, the intermediate pumps must be started before starting the primary pumps per administrative procedures. Finally, primary pumps are started sequentially. There are more than 10 separate actions, carried out at three or more locations, required to start the primary pumps. This in itself practically precludes an inadvertent pump startup during refueling operation so that the composite event would be Extremely Unlikely. However, for convenience of presentation, the event is treated in this section. Assuming the pumps are accidentally started, the full pump head would be imposed across the receptacle and across the 3/4 inch conduit connecting the high pressure region to the low pressure region, i.e., the pressure buildup would be in the direction of providing hydraulic balance. For a fully loaded core, as one would expect, an accidental full flow pump startup would not adversely effect the hydraulic balance system. If either blanket or fuel assemblies were removed from the core at the time of an accidental pump startup, the hydraulic balance could be adversely affected depending upon the number of assemblies removed. For the case of one fuel assembly removed, the reduction in hydraulic balance pressure differential would be dependent upon the relative loss in pressure due to flow through the 3/4 inch receptacle conduit as compared to the loss in pressure due to flow through the conduit connecting the low pressure manifold to the outlet plenum. The pressure in the low pressure manifold and in the conduit connecting the manifold to outlet pressure would rise quite significantly as the number of fuel assemblies are removed. A pressure differential of approximately 45 psi is necessary to float a fuel assembly and more than twelve fuel assemblies would have to be removed in order to lose hydraulic balance at full flow. It is thus concluded that the loss of hydraulic balance to the reactor fuel/blanket/control assemblies during refueling with an accidental pump startup is remote. Also, since refueling is conducted only with all the control rods fully inserted, inadvertent floating of fuel assemblies would impose no reactivity incident and the only consequence would be possible damage to the assembly. However, when the driveline is removed from a secondary control assembly during refueling, a pressure differential of 7 to 10 psi is required to cause the secondary control rods to float up 36"; more than 3 core assemblies would have to be removed to achieve this with an inadvertent pump startup. [Note: Even with all the secondary rods out, the reactor would stay subcritical due to the higher worth of the unaffected primary control rods.] Floating of blanket/fuel/control assemblies under these conditions will be prevented by limiting the number of assemblies that can be out of the core during refueling.

Pressure Surge Associated with a Check Valve Slam

Check valves will be used in CRBRP to prevent flow reversal in the heat transport loops under certain transient conditions. These create a pressure surge at closing. If the surge enters the low pressure region it will act on the fuel assemblies as a lifting force. If the pressure surge is of sufficient strength and duration it could overcome the hydraulic balance and lift the assembly. The pressure surge travels at sonic velocity (7800 ft/sec) and in the case of the 14 ft fuel assembly the maximum time the pressure surge can exert a lifting force on the assembly is 0.0018 second. Based on fast reactor design experience in which the maximum expected pressure surge is conservatively estimated to be approximately 50 psi, this pressure wave of 0.0018 second duration would lift an assembly an amount of approximately 0.001 inch which is not considered significant. This is a very conservative estimate and the reason the upward displacement is small is the extremely short duration that the pressure wave acts on the assembly. Therefore it is expected that a check valve slam will not adversely effect the hydraulic balance system.

Blockage of Conduit Providing Communication Between the Reactor Low and High Pressure Region

The most likely cause of loss of hydraulic balance to fuel/blanket/control assemblies that can be postulated is an increase in pressure in the low pressure region below the fuel/blanket/control assembly nozzles. This can occur by complete or partial blockage of the 3/4 inch and 2 inch conduits connecting the receptacle with the low pressure plenum or between the low pressure and outlet plenum. A pressure increase in the low pressure plenum can be discounted since it is connected to the outlet plenum by a large area and many flow passages. Only blockage of the 3/4 and 2 inch conduits below the assembly nozzle could cause pressure below the assembly to buildup. Specifically, if a large blockage occurred in the conduit, a significant portion of the reactor ΔP would be taken across the blockage thus raising the pressure at the bottom of the fuel/blanket/control assemblies nozzles. Several conditions can be hypothesized which could cause blockage of the conduits to cause loss of hydraulic balance. One assumes that corrosion/erosion products would plate out uniformly on the conduit inside surfaces and another assumes that large particles lodge in the conduit. It was estimated that the 0.75 inch conduit would have to be reduced to a diameter of less than 0.25 inch before an assembly would lift. Assuming a 30 year life of the core support structure, the deposition would have to exceed 10 mils per year which is extremely unlikely. The CRBRP will have a strainer (Figure 15.2.2.1-1) at the entrance to the receptacle region containing 1/4 inch holes on a pitch to diameter ratio of approximately 1.5. With a 0.75 inch conduit, the strainer will preclude the possibility of particles of sufficient size entering the receptacle and blocking the conduit. In addition the piston ring between the nozzle and receptacle will filter out particles and prevent them entering the conduit. Thus, clogging of any of the passages/conduits to cause a loss of hydraulic balance is inconceivable.

15.2.2.1.2 Analysis of Effects and Consequences

Although the loss of hydraulic balance is an unlikely event, analyses were performed to show the consequences of its occurrence. Since the highest power fuel assembly occurs at the beginning-of-equilibrium cycle (BOEC), the transients were analyzed for this particular worst period in core life. Justification for the use of this "worst" situation is found in 15.2.1.4.2.
A minimum value is obtained by decreasing the nominal Doppler coefficient 25
by 20% for uncertainties as discussed in Section 4.3.2.3. For cases that would cause reactor scram, studies were performed for both primary and secondary control rod shutdown. Scram was taken at 15% overpower with the primary control rods and at a power-to-flow ratio of 1.30 for the secondary control rods. For both systems, the maximum worth control assembly was assumed to be stuck and the shutdown worth was decreased by the appropriate amount.

In the analysis, the loss of hydraulic holddown was assumed to result in the axial displacement of a few of the 198 fuel assemblies relative to the rest of the core. The maximum amount of motion at hot, full power BOL and EOL conditions is approximately 1.8 inches and 1.2 inch, respectively. At
40 shutdown conditions, a maximum movement of 2.5 inches can occur. Axial displacements are limited to these values by the upper internals structure which acts as a mechanical secondary holddown. The effect of moving one
40 assembly, expressed in cents/inch, is given in Table 15.2.2.1-1. As can be seen in the table, the worth of the displacements decreases as one moves from the center core rows to the outer core rows. In fact, the ¢/inch

value for a Row 9 assembly is only one-tenth of that for a Row 2 assembly. The data in the table is based on the worth of the displaced fuel. If clusters of fuel assemblies move together, other reactivity effects such as the relative insertion of the control assembly would need to be taken into account. As many as 30 fuel assemblies scattered throughout the core can be affected without any two such assemblies being adjacent to one another. Thus, this is considered to be the maximum limit of applicability of the data.

If one conservatively assumed that 30 of the highest worth assemblies scattered throughout the core lost hydraulic balance instantaneously no
40 more than a 22.5¢ step could occur (taking a 2.5 inch displacement and 0.3¢/inch worth for each assembly). Analyses of various size step insertions have been performed with the FORE-II computer code (see Appendix A). The results of a parametric range of step reactivity insertions are shown in Figure 15.2.2.1-2. As can be seen, a 22.5¢ step would produce fuel assembly hot pin maximum cladding
40 temperatures of approximately 1420°F and 1430°F for primary and secondary scram, respectively. In actuality, when hydraulic holddown is lost at power the fuel assemblies would move upward which would cause a negative step since the movement is such that the net effect is an insertion of the control rods. This would cause a reactor power decrease. To achieve a positive step one source would be to assume that hydraulic balance is regained (while at full
40 power) and that the assemblies which had been initially raised 2.5 inches instantaneously fall back into position.

For small step insertions of less than 10¢, a discussion of the transient effects is given in Section 15.2.1.4. The results shown on Figure 15.2.2.1-2 for this range do not result in reactor scram (i.e., the reactor power does not reach 15% overpower conditions). As discussed in Section 15.2.1.4 the temperatures would not attain values as high as indicated on the figure since the reactor automatic control system would turn the transient around and bring the power to its original level before the temperatures would be reached. Also, there is sufficient time with small step insertions for the operator to manually restore the reactor to full power.

Similar analyses to those described above have been performed for the reactor at hot standby conditions (600°F reactor inlet and subcritical power generation) and it was found that the resultant maximum cladding temperatures are significantly less than those given by Figure 15.2.2.1-2 for the same size reactivity insertion. Also, full power and hot standby runs have been made for the highest power radial blanket assembly hot pin and the resultant temperatures are significantly less than those for the fuel assembly hot pin at full power.

As indicated in Section 15.1. parametrics were performed to show the effect of using "minimum required" primary control rod shutdown rate values instead of the "expected" values (both having the highest worth rod assumed to be stuck). The temperatures described thus far in this section have been based on the expected rates of shutdown worth which as described in this

earlier section are felt to give the more realistic evaluation of the transient. Figure 15.2.2.1-3 shows the fuel assembly hot spot cladding temperature for the two cases. As can be seen, in the range of interest for the loss of hydraulic holddown event (i.e., a 22.5¢ step reactivity insertion) there is
40 an insignificant increase of about 5°F due to the differences in shutdown rate. Also, as can be seen on the figure, even for an extremely large step of 90¢, the increase would only be about 60°F.

15.2.2.1.3 Conclusions

Although the loss of hydraulic holddown is an unlikely event, conservative analyses were performed to determine the effect of this transient on the core. For the postulated case of 30 fuel assemblies scattered throughout the core axially moving 2.5 inches, the maximum step reactivity insertion to the core was found to be less than 22.5¢. For primary and secondary scram, this
40 would cause maximum cladding temperatures of 1420°F and 1430°F, respectively. These transient temperatures should not result in any significant degradation of pin lifetime. A description of how this type event is incorporated in the pin cladding structural design evaluation is given in Section 4.2.1.3.

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TABLE 15.2.2.1-1

THE REACTIVITY EFFECT OF THE AXIAL MOTION OF A FUEL ASSEMBLY

Row Number	Cents/inch for 1 Fuel Assembly*
2	0.30
3	0.27
4	0.25
5	0.22
6	0.19
7	0.12
8	0.07
9	0.03

*The reactivity effect is negative when the fuel assembly moves upward (out of the core) and positive when it moves down (into the core).

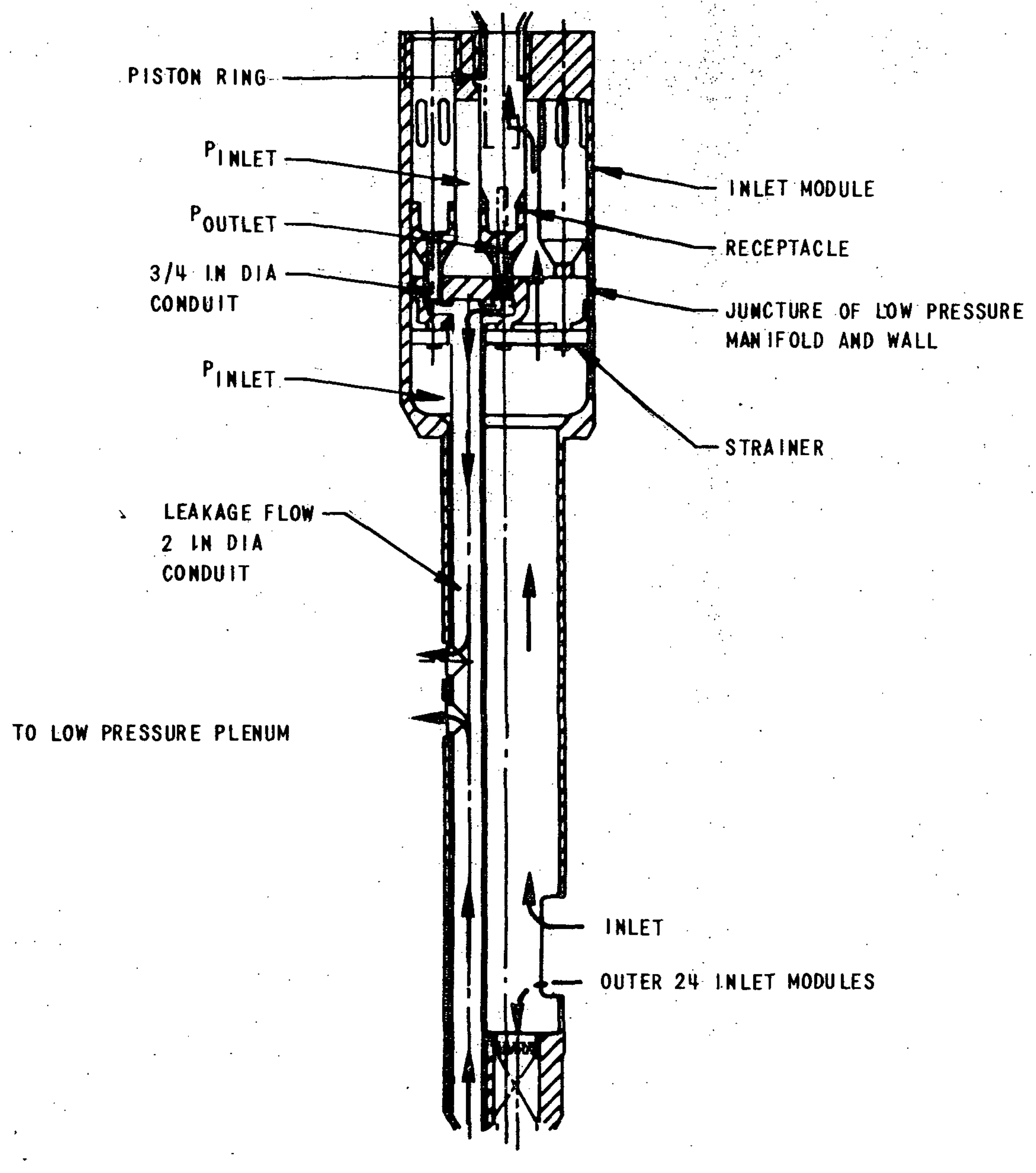

Figure 15.2.2.1-1. Schematic of Hydraulic Holddown Design

6727-23

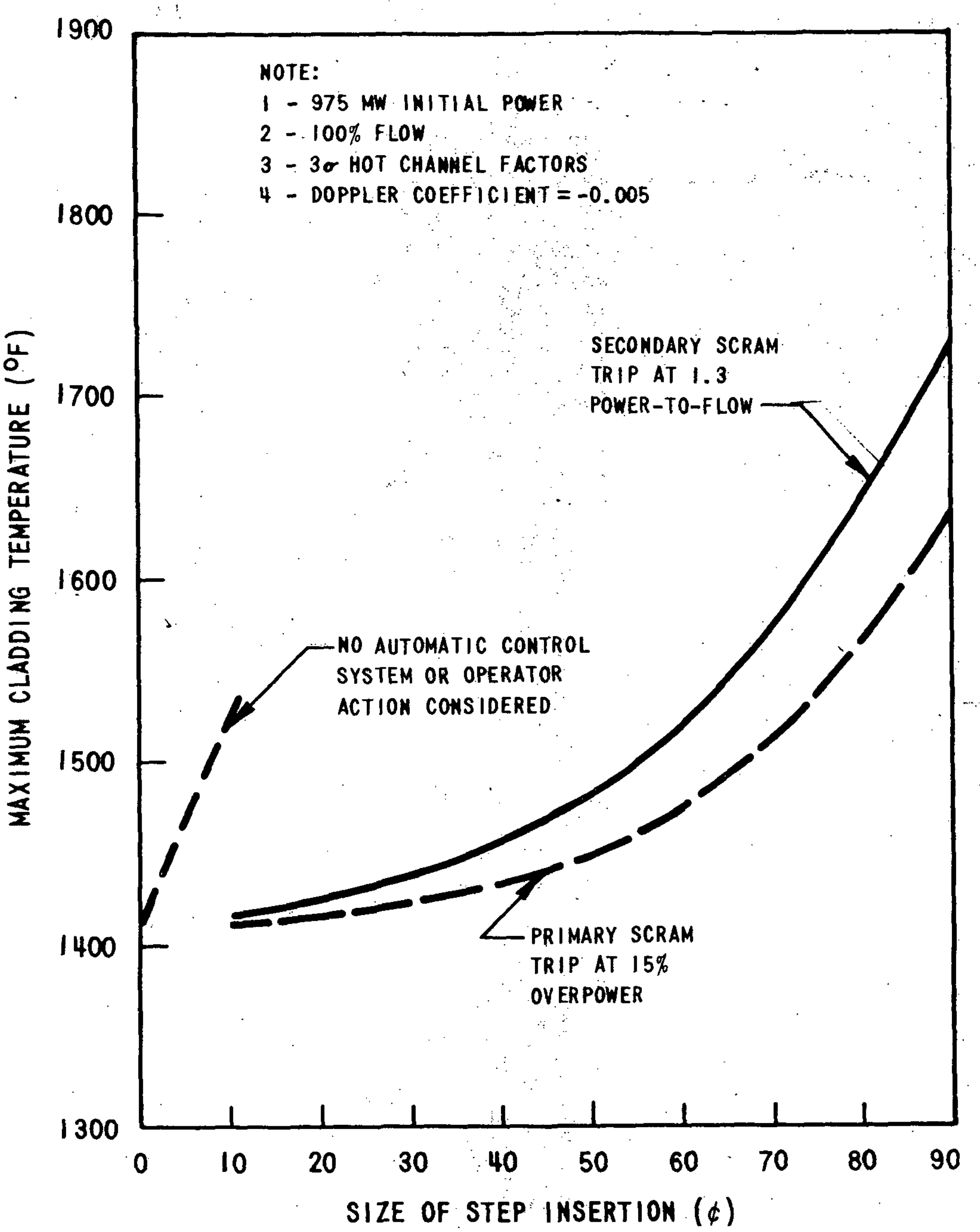

Figure 15.2.2.1-2. Effect of Various Size Step Reactivity Insertions on F/A Maximum Cladding Temperature

6727-24

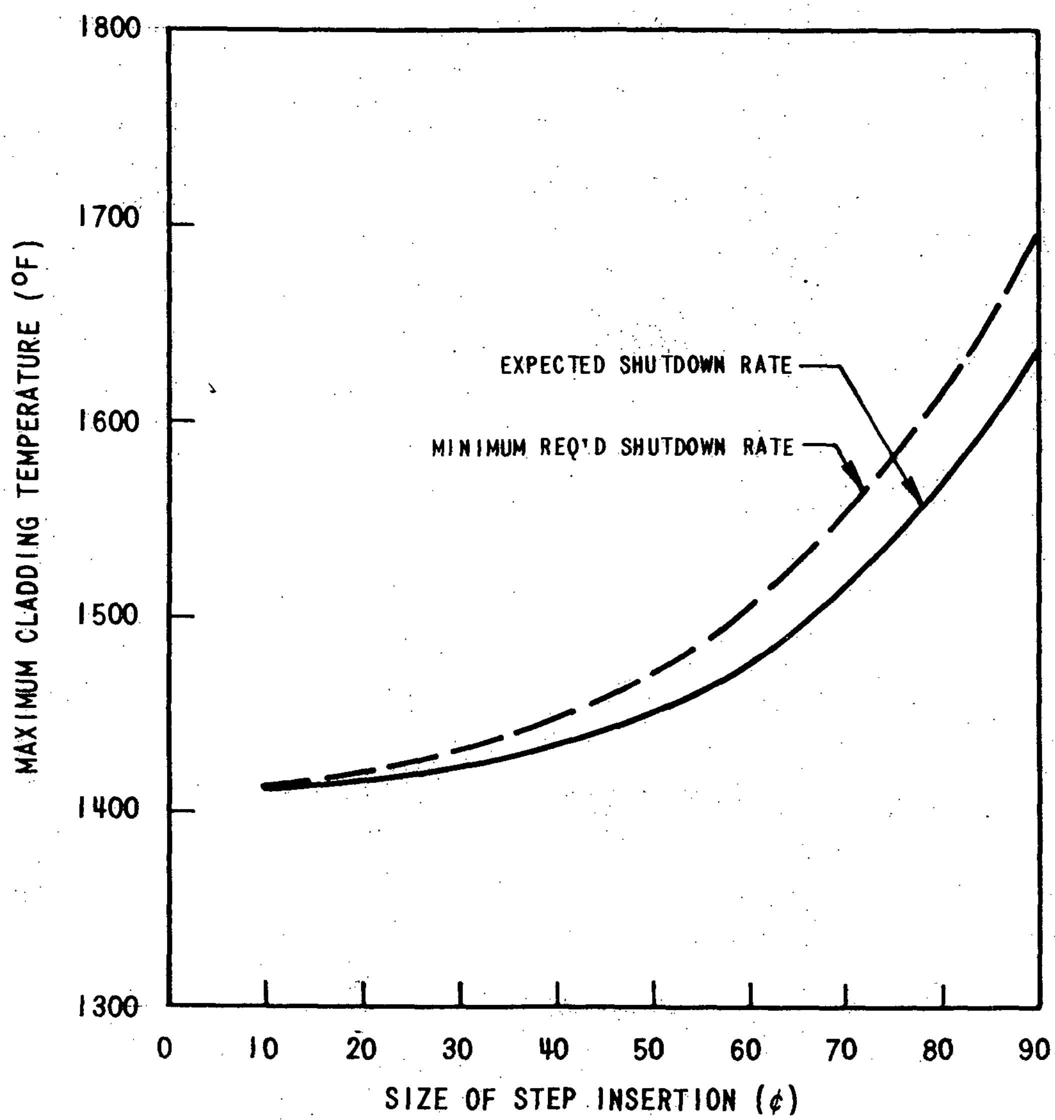

Figure 15.2.2.1-3. **Comparison of F/A Hot Spot Cladding Temperature Using Expected and Minimum Required Primary Control Rod Shutdown Rates**

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15.2.2.2 Sudden Core Radial Movement

15.2.2.2.1 Identification of Causes and Accident Description

The event to be considered here involves core radial motion which occurs rapidly and is difficult to accurately predict. This is in contrast to normal core radial motion which occurs gradually and predictably in response to normal temperature changes and irradiation induced material swelling and creep. The latter type event is discussed in Secton 4.2.2.4.3.

The type of sudden core radial motion to be evaluated has been termed "stick-slip" motion. Stick-slip motion refers to a situation in which the reactor assemblies are restrained from moving radially by interassembly frictional forces at the assembly load planes (stick) and then suddenly move to a new position dictated by current temperature and irradiation environment as the interassembly frictional forces are suddenly removed or reduced (slip). If it is postulated that sticking occurs while the reactor assemblies are bowed away from the core centerline, a sudden positive reactivity insertion can take place as the assemblies sip to an inwardly bowed shape (towards the core centerline). Such an event is unlikely since the buildup of interassembly frictional forces which would be required to cause sticking would occur only when the assemblies are in a compact inwardly bowed state. If the assemblies are bowed outward away from the core centerline, the interassembly gaps would be larger and then the probability of sticking would be minimal. On the other hand, if because of themal and irradiation effects the assemblies due to manufacturing tolerances and frictional forces.

If the assemblies are prevented from achieving a compact state due to interassembly frictional effects, it is possible that a seismic event could overcome the frictional effects and allow the reactor assemblies to take on a more compact state. This is considered to be the only realistic initiating mechanism for a stick-slip type event.

If the stick-slip event occurred, the reactivity insertion would cause temperature rises of the fuel, cladding, and coolant. The power rise would trigger a primary control system scram if the limits of Section 15.1.3 were exceeded.

15.2.2.2.2 Event Evaluation: Model, Assumptions, and Conservatisms

To determine the maximum possible reactivity insertion, the following analysis steps were followed:

1. Predict the difference in core assembly positions and bowing between refueling and full power.

2. Determine the reactivity worth factors associated with radial motion of each core assembly.

3. From the predictions of maximum possible radial motion and worth factors, determine an upper limit for possible reactivity insertion from stick-slip.

To predict the core assembly positions and bowing at refueling and full power conditions, a finite element model was constructed of a radial row of core assemblies. The reactor environmental conditions were then applied along with material characteristics to give bowing and position curves like those of Figures 4.2-88 through 4.2-92. Refer to Section 4.2.2.4.3 for further details of the core assembly bowing analysis. Comparison of the bowing shapes for Figures 4.2-88 through 4.2-92 shows an inward bowing at full power (100% power to flow ratio). The reactor assemblies were assumed to stick in the refueling position (at 0% power to flow ratio) and to then slip suddenly to the full power position.

Conservative nominal compaction reactivity worth coefficients were determined by using the assumptions that all control rods would be parked above the core at the beginning of an equilibirum cycle. The worth coefficients are shown in Table 4.3-14.

The above procedure results in a prediction of approximately 60 for the maximum value of step reactivity insertion (see Section 4.2.2.4.3).

The above upper limit is considered to be conservative for the following reasons:

1. In the analysis, all the gaps in the core were compressed compleely out whereas core compaction tests (1) indicate that not all gaps will be compressed out in a real core. This is due to manufacturing tolerances as well as frictional effects in the core.

2. The analysis assumptions were that sticking of the core assemblies would occur where the assemblies are in their maximum outwardly bowed configuration. More realistically the sticking would not occur until substantial inwardly directed thermal bowing had already occurred and forces had begun to build-up between assemblies. Thus, part of the bowing reactivity change can be expected to occur gradually which will be compensated for by Doppler and thermal expansion effects. This would reduce the maximum possible step reactivity change.

3. The inherent vibrational motion of the core assemblies when flow is passing through would tend to prevent sticking. This would aid in allowing smooth translation of the core assemblies in response to thermal bowing.

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1. W. C. Kinsel, "FTR Core Compaction and Withdrawal Tests," May 1973, HEDL-TME-73-58, UC-79 e, g, h.

4. The compliance of the load pads was modeled for two-face loading rather than multi-face loading. This provides more conservatism since it predicts more translation of the core assembly positions from refueling conditions to 100% power conditions than if multi-face loadings were assumed. This is due to more flexibility the load pads under two-face loading.

5. Temperature gradients used in the thermal bowing analysis are conservative as described in Section 4.2.2. Thus, the bowing magnitudes are overpredicted.

15.2.2.2.3 Analysis of Effects and Consequences

The limiting criteria for this event is fuel cladding temperature as a function of time. Figure 15.2.2.2-1 shows the three sigma confidence level cladding temperatures for 30¢ and 60¢ reactivity insertions and subsequent scram of the primary control rods at their normal scram rate. The cladding temperature associated with a 60¢ step reactivity insertion is about 1470°F and peaks in about 0.6 seconds.

If a secondary scram were relied upon rather than a primary scram, the maximum cladding temperature would go to 1510°F (see Section 15.2.2.1).

If a seismic event were to initiate the stick-slip event, the control rod scram response time would be somewhat slower. However, Figure 15.2.3.3-5 of Section 15.2.3.3 shows that even under these conditions the three sigma confidence level maximum cladding temperature is about 1510°F for a 60¢ reactivity insertion under SSE faulted conditions. The temperature begins to drop after about 1.2 seconds.

Maximum cladding temperature is related to cladding strain limits in Section 4.2.1.3.1.1. Figure 4.2-20 presents the umbrella transient for the fuel rod cladding emergency event. The umbrella transient is started at the two sigma confidence level maximum cladding temperature but the peak value of 1590°F represents a three sigma maximum to be included in the transient umbrella. The transient umbrella includes temperatures above 1510°F for a duration of about 34 seconds.

As cladding temperature rises during a transient event, the fission gas plenum pressure increases and the cladding stress and strain increase. In addition, the material strength is decreasing as the cladding temperature rises. Transient events where the cladding temperature remains at elevated temperatures for substantial lengths of time (10 to 20 seconds) are thus more limiting than rapid transients such as step insertions where a comparable temperature remains critically high for periods of about 2 seconds or less.

Thus, the cladding temperature effects of a 60¢ step insertion are within the emergency transient umbrella which gives acceptable transient strains as described in Section 4.2.1.3.1.1.

15.2.2.2.4 <u>Conclusion</u>

A conservative analysis was performed for an unlikely stick-slip event over the entire range of bowing transition corresponding to refueling and full power operating conditions. It was found that a step reactivity insertion of approximately 60¢ was predicted as an upper limit.

For a 60¢ step insertion maximum cladding temperature would not exceed 1470°F for normal control system SCRAM conditions and would not exceed 1510°F under SSE conditions. This compares with an allowable temperature of less than 1600°F for emergency conditions (unlikely events).

Thus, no significant degradation of the cladding would be expected for a temperature of this magnitude and duration.

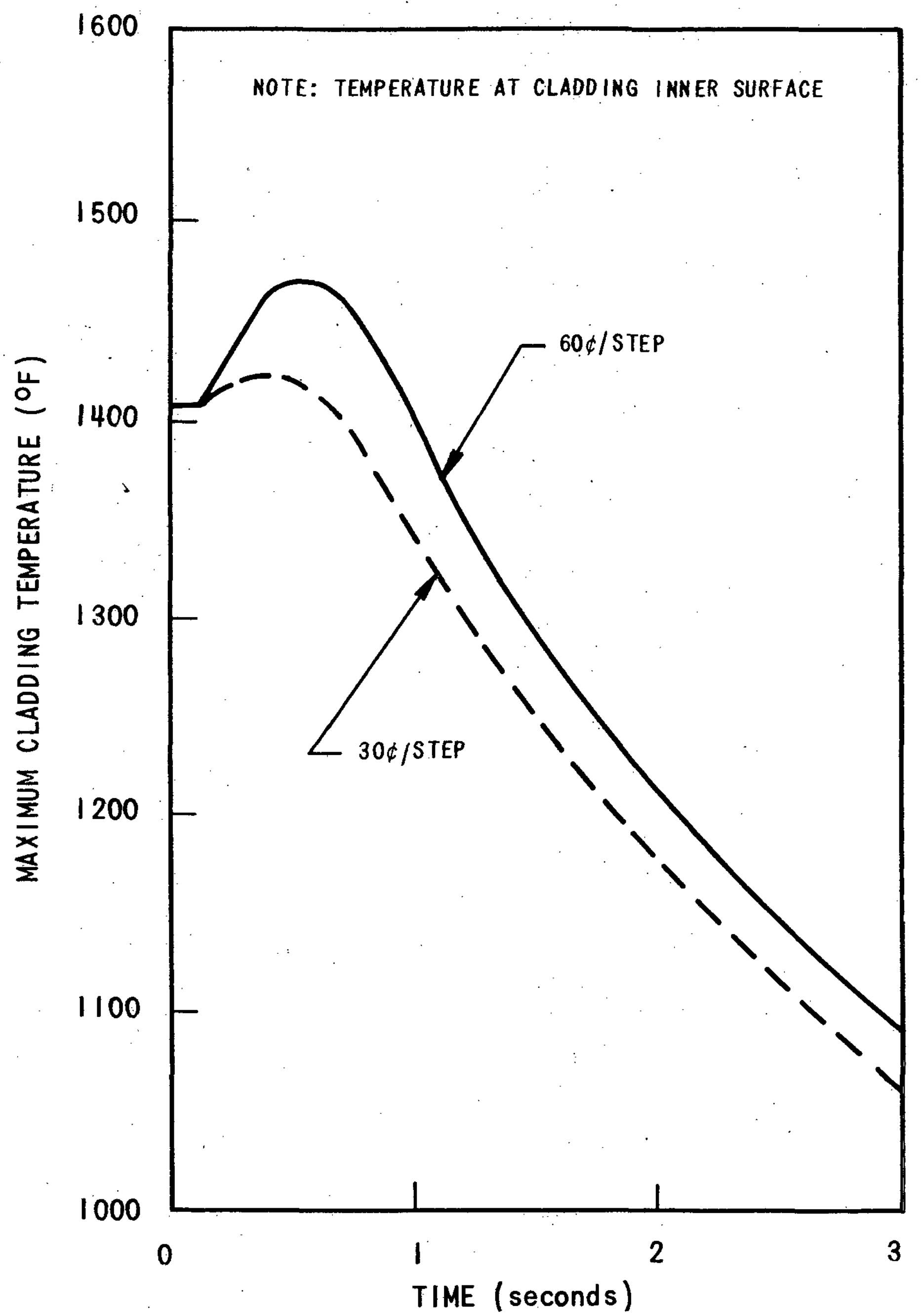

Figure 15.2.2.2-1. Variation of Maximum Cladding Temperature; 100% Power/ Flow, Primary Shutdown; 975 MWt Initial Power

6727-6

15.2.2.3 Maloperation of Reactor Plant Controllers

15.2.2.3.1 Identification of Causes and Accident Description

During normal operation over the load range, the automatic control system varies reactor power to maintain system temperatures at the desired values. The block diagrams of the overall control system, the reactor control, and the control rod drive mechanism controllers are included and described in Section 7.7. Postulated failures in this equipment would result in power excursions. Depending on the nature of the failures postulated, the excursion would be self-limiting, limited by action of control rod withdrawal blocks, or terminated by the shutdown system. These failures can be effectively evaluated by considering the equipment starting with the CRDM controllers and working through the remainder of the control equipment. Figure 15.2.2.3-1 is a simplified block diagram of the key elements. Single failures of control equipment are defined as anticipated, multiple failures of independent elements are defined as Unlikely.

First, failures in the CRDM control programmer could result in unwarranted rod withdrawal. Single failures can be postulated which would result in withdrawal at 9 inches per minute or less. The speed limiter circuitry acts to positively block rod motion for pulse rates corresponding to speeds greater than 9 inches per minute. Single rod withdrawal at 9 inches per minute or less results in a reactivity insertion of less than 3¢/second even if the highest worth rod is assumed at its highest differential worth position. If the control system is in normal automatic mode, the other rods will move in to compensate for the excursion and no transient results. If the control system is assumed inoperative, the rod blocks will terminate motion prior to scram for many of the postulated CRDM controller failures. If neither the control system nor the rod blocks operate, the scram system terminates the results of a 3¢/second withdrawal without exceeding the operational incident limits, as discussed in Section 15.2.1.2.

Multiple simultaneous failures within a CRDM controller or failure involving multiple CRDM controllers simultaneously are highly improbable. Regardless of the number of CRDM controller failures, a CRDM is incapable
51 | of maintaining outward motion at rates in excess of 73 inches per minute. At this rate, the centrifugal force on the collapsible rotor arms plus the dead weight and dynamic forces overcomes the magnetic insertion force and the rod releases. Therefore, the maximum reactivity insertion involving a single rod is less than 20¢/second. While this failure is highly improbable, it is nevertheless terminated by the PPS without exceeding the minor incident unlikely limits as necessary, and in fact, the results do not exceed the operational incident level as discussed in Section 15.2.3.5.

Failures of multiple individual CRDM controllers and postulated failure of the sequences produce the same effect. Several rods are withdrawn in the staggered steps characteristic of sequence operation. Since the control system is rendered inoperative by the postulated sequencer failure, no remedial action is available through normal control. However, the rod blocks function independently of the sequences and limit the magnitude of the excursion by stopping outward motion. If the additional failures of the rod block circuits are postulated, the sequences are rate limited to prevent withdrawals corresponding to 5¢/second or more from the banked rods. Therefore, single failures of the sequencer plus failure of the rod blocks result in an excursion limited to 5¢/second which is terminated by the PPS within the limits of an operational incident.

Postulating the concurrent failures of the sequencer and its limits circuits and the rod blocks results in 7 rods moving out but limited to 9 inches per minute withdrawal rate. This corresponds to less than 20¢/second and is terminated by the PPS within the minor incident limits (actually with the operational incident limits). Additional postulated failures are not physically realistic since four independent failures have already been postulated.

Failures within the controller equipment in front of the sequencer cannot result in transients more severe than the multiple failure case specified above. The speed limits and rod blocks which are independent of the control equipment provide assurance that unlimited withdrawals do not occur as a result of single failures within the system.

15.2.2.3.2 Analysis of Effects and Consequences

The analysis of the 5¢/second ramp insertion is described in Section 15.2.1.4. Based on these results, the first line protection terminates the transient within the operational incident limit. The analysis of the 20¢/second ramp insertion is described in Section 15.2.3.5. The results show that the transient is terminated within the minor incident limits.

15.2.2.3.3 Conclusion

Multiple independent failures of control equipment do not cause reactivity insertions larger than 20¢/second. These insertions are terminated within appropriate limits by PPS action as shown in 15.2.3.5.

6727-26

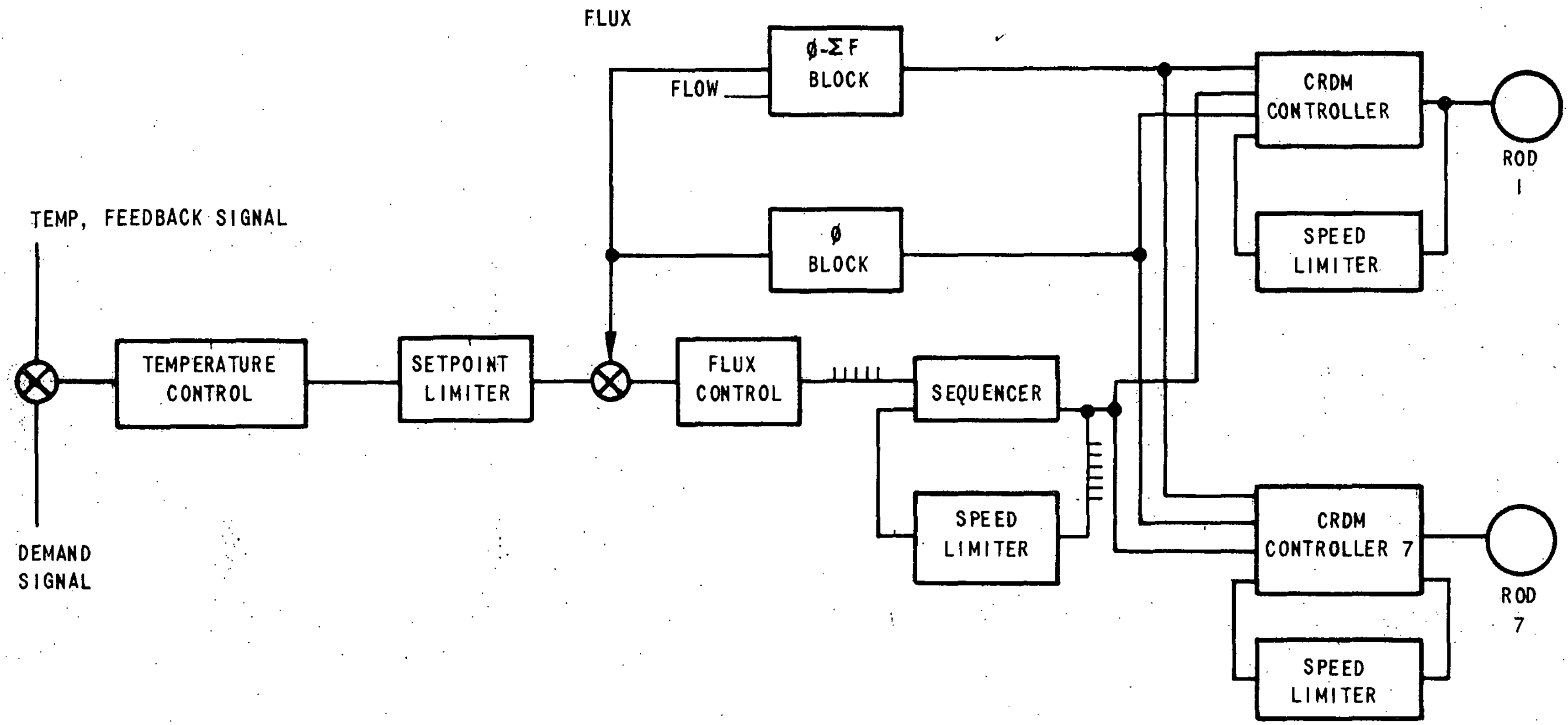

Figure 15.2.2.3-1. Block Diagram of Key Elements of Control Equipment

15.2.3 Extremely Unlikely Events

15.2.3.1 Cold Sodium Insertion

15.2.3.1.1 Identification of Causes and Accident Description

Sodium is supplied to the reactor vessel either by the coolant makeup system or the three primary loops. The reactor coolant make-up system removes and returns sodium to the outlet plenum. It can be operated either as a make-up flow system or as an overflow heat removal system (DHRS). In the make up flow operating mode the inlet and outlet temperatures of the sodium are approximately the same. However, in an DHRS operating mode the return temperature could be approximately 350°F less than the overflow coolant tem-
51 perature (see Section 5.6.2 for details). The DHRS is only operated during shutdown (with decay heat generation in the core) with the control rods fully inserted. The insertion of cold sodium into the core with control rods fully in does not institute a reactivity insertion problem.

With respect to the primary loops, the accidental startup of an inactive loop while on two loop operation is the only potential source for introducing a significant amount of cold fluid suddenly into the reactor vessel lower plenum. If one loop is not operating, it could contain sodium with its temperature as low as 400°F. Flow of this sodium into the reactor could cause a substantial reduction in the average core coolant temperature.

The startup of an inactive loop is prevented by administrative procedures, interlocks, and PPS functions. Only three sequences for startup of the inactive loop can be identified: (1) the primary pump is incorrectly started; (2) the intermediate pump is incorrectly started followed by an attempted incorrect start of the primary pump; or (3) both pumps are simultaneously started. In any case, the PPS acts to shut the plant down.

For the case where a primary pump is incorrectly started while the plant is operating at power on two loops, the primary to intermediate speed ratio trip subsystem causes plant shutdown as soon as the primary pump reaches a predetermined speed. Assuming the other loops are operating at 100% flow, the check valve on the shutdown loop remains closed (for the short time interval before the trip signal occurs) due to the high pressure on the downstream side and no cold sodium can be inserted. Furthermore, the primary to intermediate flow ratio subsystem initiates a trip when the primary flow in the shutdown loop reaches a fraction of full flow and provides an independent back-up to the trip. Even if flow is initiated in the shutdown loop, the rapidity of the trip and the system time constants prevent significant cold sodium from reaching the core inlet prior to scram. In both the speed and flow rate trip subsystems the startup permissive is blocked, thereby activating the trip when the reactor is above 5% full power. If an intermediate pump is started, both of the aforementioned subsystems cause a trip at approximately the same time. However, no cold sodium insertion can occur because the primary pump in the shutdown loop has not been started.

Deliberate operator coordination, which violates several administrative procedures, is required to start both the primary and the intermediate pumps of the shutdown loop simultaneously. The resulting flow and pump speed increases are not identical in the two loops because the primary pump must operate against the back pressure of the two operating loops. Consequently, the primary to intermediate speed ratio subsystems operate to produce reactor shutdown and thus prevent the cold sodium insertion.

15.2.3.1.2 Analysis of Effects and Consequences

To demonstrate the inherent safety margins available, the effect of cold sodium insertion to the core was studied. It must be emphasized that in actuality PPS action would occur (as described earlier) preventing the accident. Initial two loop at two-thirds full operating power and flow was assumed.

The shutdown loop was considered to contribute one-third of the total flow when it is started. Since the highest power fuel assembly occurs at the beginning-of-equilibrium (BOEC), the transient was analyzed for this worst period in core life. Justification for the use of this "worst" situation is found in 15.2.1.4.2. For scram the highest worth rod was assumed to be stuck and the shutdown worth of the control system was reduced by the appropriate amount. The primary control rod system was assumed to trip at 115% operating power.* The secondary system trip would be at a 1.30 power-to-flow ratio. As will be shown in the ensuing discussion, the power does not increase sufficiently to reach this secondary trip level. 25

As the temperature of the sodium coolant entering the reactor core decreases, the resulting higher density would produce an increase in reactivity. Also, since the fuel temperatures decrease initially, an additional reactivity increase results from the effects of the Doppler coefficient. In the analysis, it was assumed that the cold sodium mixes perfectly with the hot sodium in the inlet plenum. For conservatism the sodium inlet temperature was assumed to drop to 624°F (although this cannot physically occur) and remain at this value throughout the transient. Although it is physically impossible for the flow to increase instantaneously to its normal full flow value, this assumption was conservatively made for the analysis. This results in the largest rate of sodium cool-down for the core. The Doppler coefficient was taken to be its maximum value of -0.0074 which corresponds to beginning of equilibrium cycle core conditions. This results in slightly more pessimistic conditions that if one were to use the minimum Doppler coefficient that is assumed in the reactivity insertion events described in Section 15.2. The sodium density coefficient was taken to be that of beginning-of-cycle (BOC1). Since sodium density feedback is small compared to Doppler feedback (see Section 4.3.2, the effect of not using the actual BOEC values would be small. The analysis was performed using the FORE-II computer code (see Appendix A).

Figure 15.2.3.1-1 shows the behavior of the relative reactor power with time after initiation of the transient. Primary control rod system scram is observed to occur very early in the transient at approximately

*This assumes the trip is reset from its full power value when operating at reduced power.

2.6 seconds after initiation of the transient. Under this condition, the reactor reaches about 76% of full power before shutdown (initially the power is 67% of full power). Secondary control rod system scram would not occur since a power-to-flow ratio of 1.3 is not reached (after 50 seconds the ratio is about 0.9).

Figures 15.2.3.1-2 through -5 give the effect of the transient on the maximum fuel, cladding, and coolant temperatures. These temperatures are given for their respective hottest positions which are at the center of the active core for the fuel and near the top of the active core for the cladding and coolant temperatures. The maximum fuel temperature resulting from the transient is observed to remain below the normal full power operating temperature. The rise in fuel temperatures results from the increased power due to the small positive reactivity effects produced by the Doppler coefficient and the higher density of the sodium coolant.

In contrast to the slight rise in fuel temperatures, the transient produces a rapid decrease in cladding temperatures. Figure 15.2.3.1-3 shows the variation on the maximum cladding temperature at the top of the active core. As can be seen, the cladding is at its highest temperature for the transient at the initiation of the event.

The transient behavior of the maximum coolant temperature is shown in Figure 15.2.3.1-5. The behavior of the coolant temperature follows that of the cladding temperature. The coolant temperature can be observed to remain below the 1296°F value it would have for the case of steady state full power operation throughout the transient.

The time constant for the sodium to return to the fuel assembly inlet region after passing through the core, outlet and inlet plena and primary loop is on the order of 60 seconds at full flow. The sodium returning to the fuel assemblies could have an increased inlet temperature from the 624°F value. Thus, the reactor power would decrease due to the increased inlet temperature effects (via sodium density coefficient and Doppler feedbacks).

15.2.3.1.3 Conclusions

Sudden insertion of cold sodium to the CRBRP core cannot occur even if both the primary and intermediate pumps on a shutdown loop are started simultaneously, or if the intermediate pump is started followed by an attempted startup of the primary pump, or if the primary pump is incorrectly started. If startup is attempted the plant protection system acts through the primary to intermediate flow ratio and speed ratio subsystems to produce reactor shutdown.

Although scram would occur before cold sodium could enter the core, a hypothetical analysis showing core results if cold sodium could be inserted was presented to demonstrate the plant's safety margin. Core power and temperature changes were shown to be minor even without a scram occurring for 50 seconds. In fact, the cladding temperature remains at least a hundred degrees lower than its initial steady state temperature before the accident. It must be emphasized that the transient temperatures shown would not actually be incurred on the reactor due to the PPS action.

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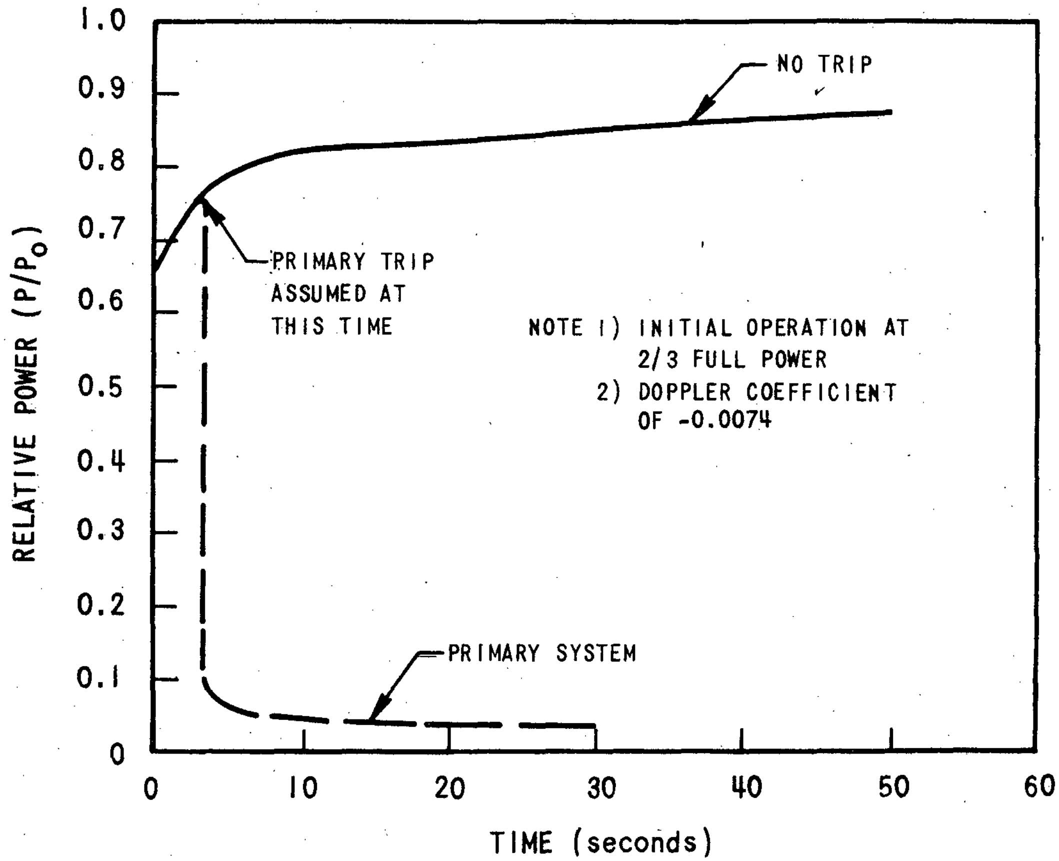

Figure 15.2.3.1-1. Relative Power Vs. Time for Cold Sodium Insertion

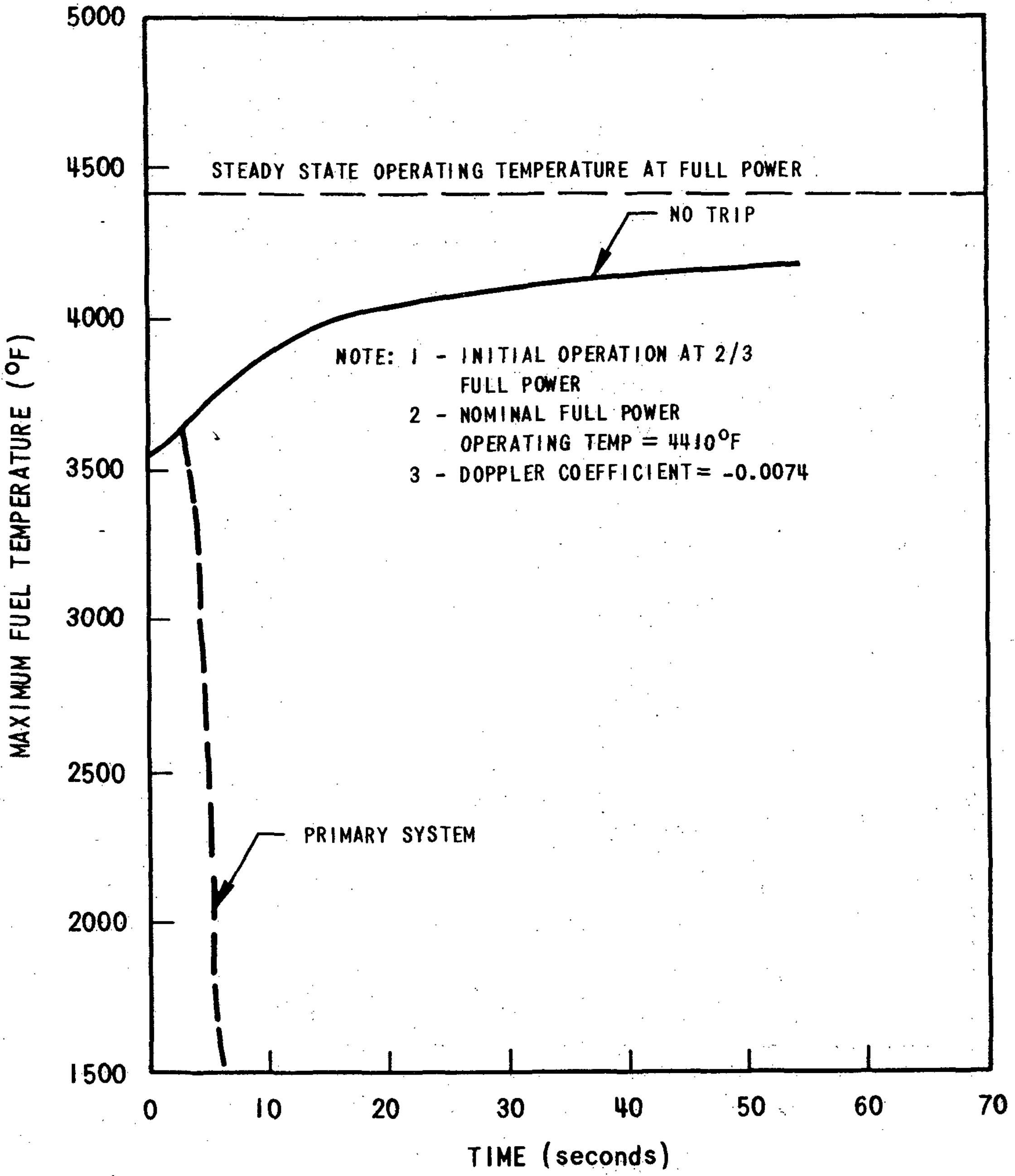

Figure 15.2.3.1-2. Maximum Fuel Temperature Vs. Time for Cold Sodium Insertion

6727-28

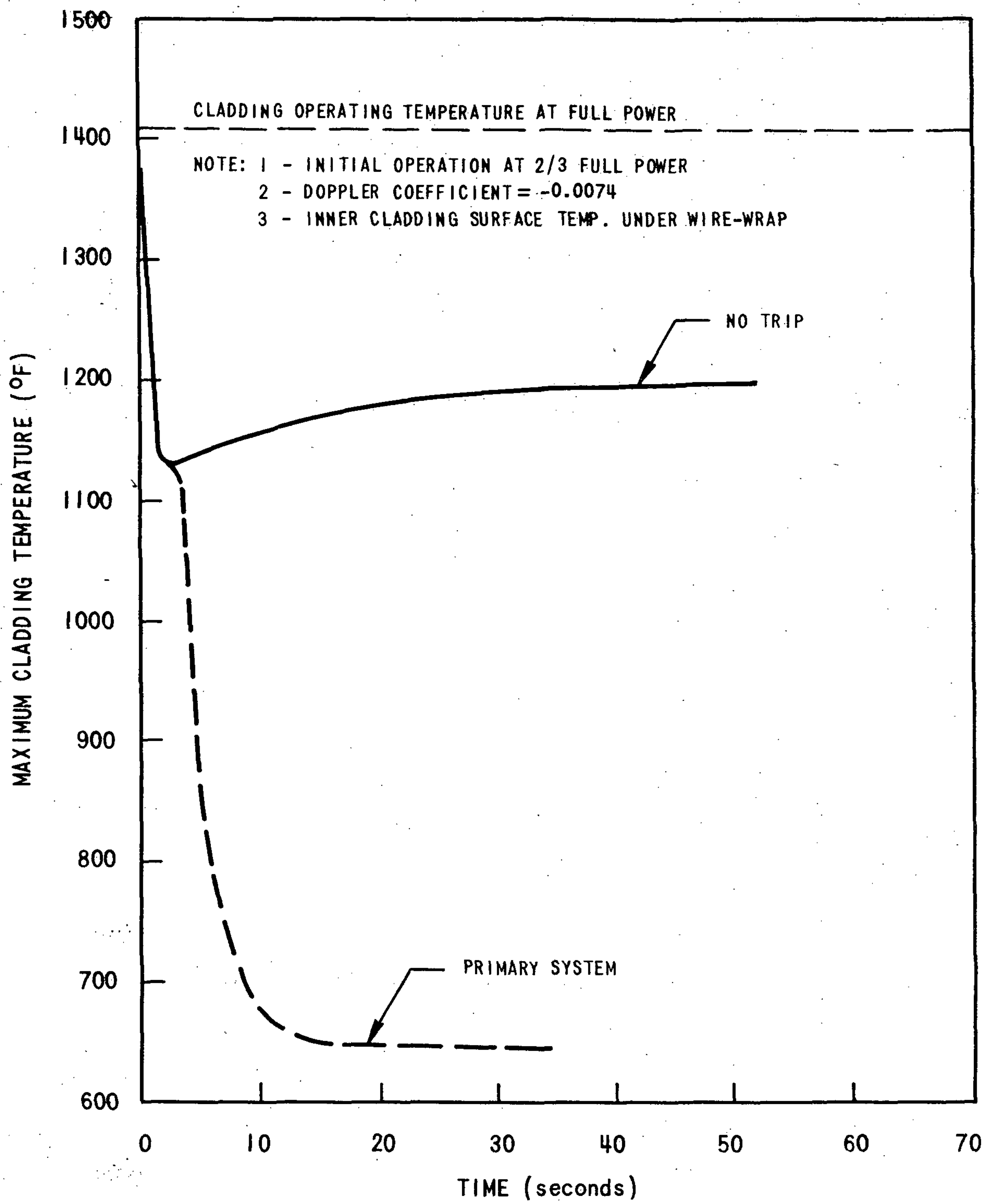

Figure 15.2.3.1-3. Maximum Cladding Temperature at Top of Active Core Vs. Time for Cold Sodium Insertion

6727-29

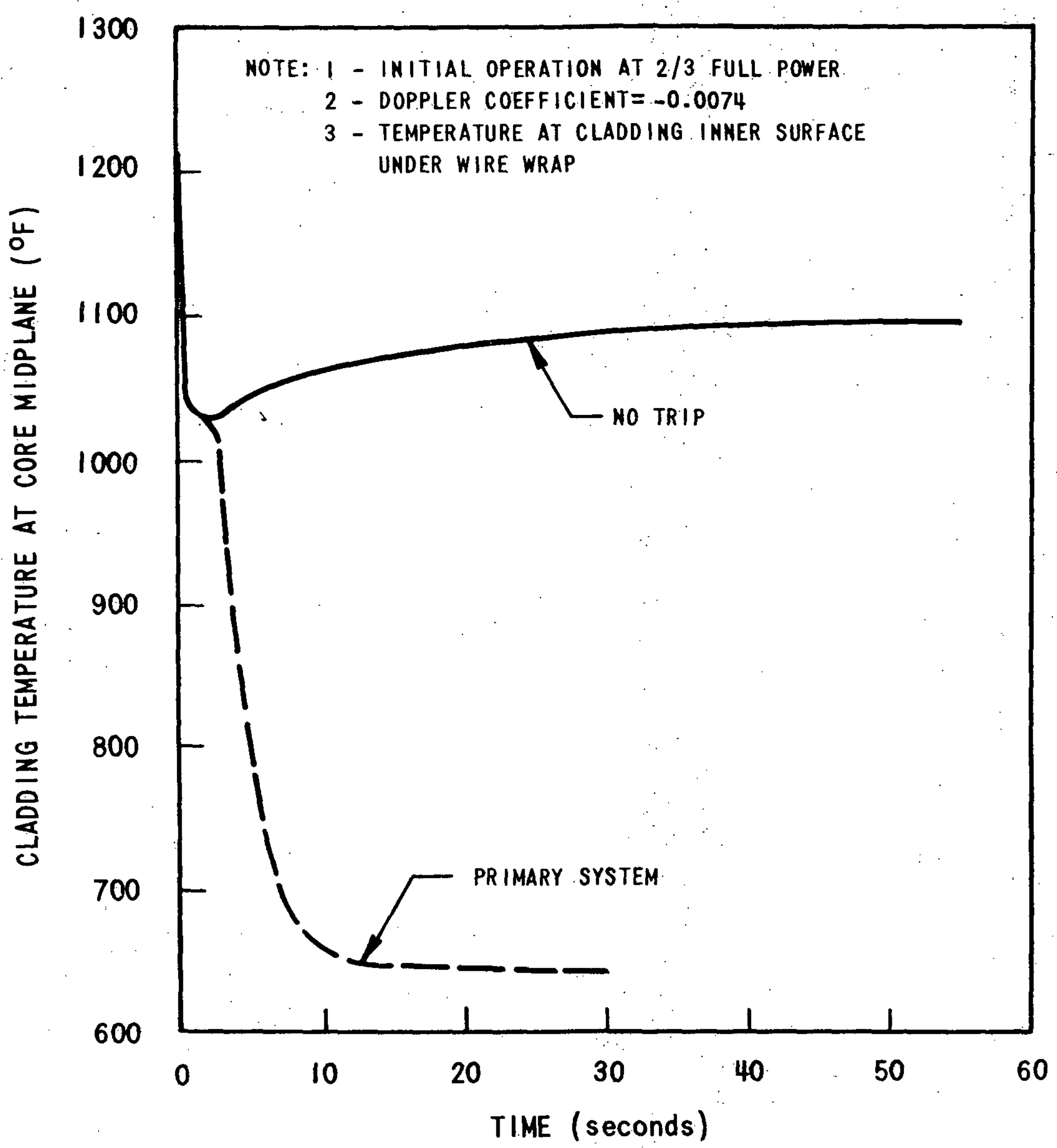

Figure 15.2.3.1-4: F/A Hot Channel Cladding Temperature (At Core Midplane) Vs. Time for Cold Sodium Insertion

6727-30

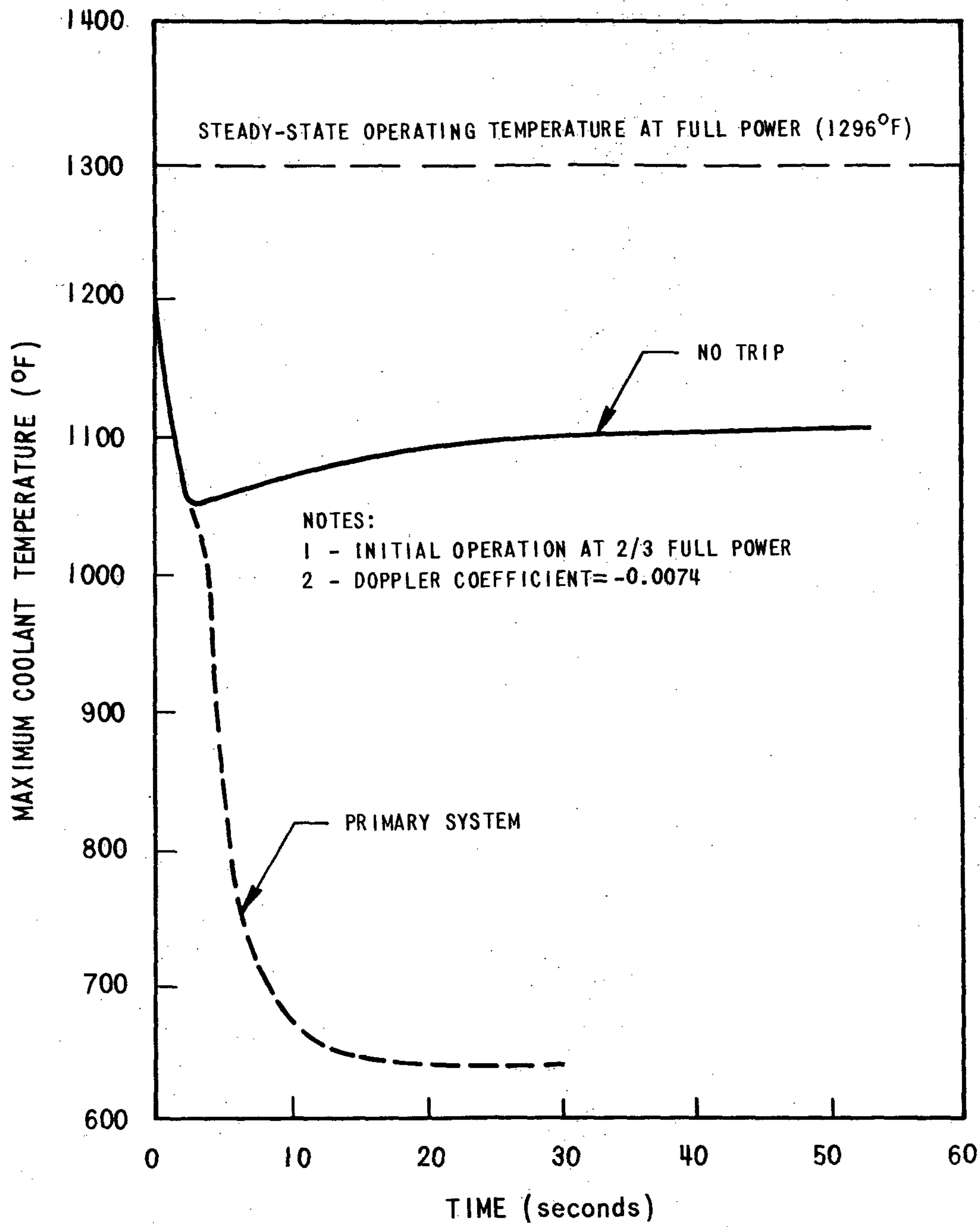

Figure 15.2.3.1-5. Hot Channel Coolant Temperature at Top of Upper Blanket Axial Position Vs. Time for Cold Sodium Insertion

6727-31

15.2.3.2 Gas Bubble Passage Through Fuel, Radial Blanket and Control Assemblies

15.2.3.2.1 Identification of Causes and Accident Description

The neutronic characteristics of the CRBRP core are such that voiding of the coolant in the central core region creates a positive reactivity effect, whereas voiding at the core periphery creates a negative reactivity effect.

The reactivity worth of the central, positive worth region is greatest at the end of the equilibrium cycle. Table 15.2.3.2-1 shows the worth of various rows of the core as a function of axial position for this time in core life. The void worths shown in this table and the uncertainties in these values are discussed in Section 4.3.2.3.2. As can be seen, the last row of fuel assemblies (Row 9) and all the radial blanket rows have a negative void worth for all positions.

Voids in portions of the core may be postulated to arise from sudden large ruptures of fuel pins releasing fission product gases, or from bubbles which originate in the heat transport system (e.g., argon cover gas) and enter the reactor. Voids from each of these sources are either extremely unlikely or inconsequential or both. As will be discussed in detail in Section 15.4, Local Failure Events, the failure of a pin releasing a large burst of gas is unlikely. Bubbles of any significant size cannot be formed in the heat transport system since entrainment and hold-up of gas in the primary system is inhibited by the design.

The heat transport system incorporates design features to preclude gas bubbles from entering the core. These include:

a. Vents provided to eliminate possible gas pockets that may form during sodium fill.

b. A low cover gas pressure which reduces gas entrainment.

c. A continuous bleed from the top of the IHX to prevent accumulation of gas during operation is provided.

d. The Primary Pump is designed to eliminate vortexing and gas entrainment.

e. A high fluid velocity in the piping between the pump discharge and the vessel inlet minimizes the possibility of gas entrainment.

f. A vortex suppressor at the optimum depth to prevent gas entrainment is located in the outlet plenum.

g. One or more holes with special pressure reducers will be provided in the core support cone to vent gas from underneath the cone.

Experiments have shown (Ref. 1) that in the FFTF, if a significant size bubble were to reach the reactor inlet plenum, turbulence would result in dispersion of the bubble into small bubbles before entry into the core and that large coherent bubbles entering the core are impossible. These small bubbles could only slightly reduce the coolant density with very little effect on reactivity. Due to the similarity in the inlet plenum design it is expected that similar results would be obtained for CRBRP; however, tests as described in Section 1.5 will be made for the CRBRP design. 9

Despite the demonstration of bubble breakup, and the design features outlined, which results in extremely low probability of a void of significant size occurring in the core, an analysis of bubble effect has been performed, covering a range of bubble sizes which are well in excess of any credible bubble size.

15.2.3.2.2 <u>Analysis of Effects and Consequences</u>

An analysis was performed to determine the effects of postulated coherent bubbles of given initial height (treated as a parameter) which occupied all of the fuel assemblies in Rows 1 to 4 or Rows 1 to 8 (also treated as a parameter). Radial sizes were conservatively assumed not to extend beyond Row 8 since this is the last row of positive void worth. The gas slug was assumed to form at the bottom of the lower axial blanket and move uniformly through the core with a speed equal to the average flow velocity. It first passes through the negative void region, then into the positive region and finally into the top negative region (see Table 15.2.3.2-1).

Since the highest power fuel assembly occurs at the beginning-of-equilibrium cycle (BOEC), the transient was analyzed for this particular worst period in core life. Justification for the use of this "worst" situation is found in 15.2.1.4.2. A minimum value Doppler coefficient of 25
-0.005 was used for this core condition. As indicated earlier, the worst case void worths are for the end of equilibrium cycle; thus, these conservatively used for the study. Analyses were performed for both primary and secondary control rod shutdown. Scram with the primary control rods was taken at 15% overpower and at a power-to-flow ratio of 1.30 for the secondary control rods. For both systems, the maximum worth control assembly was assumed to be stuck and the shutdown worth was decreased by the appro-
52 priate amount. Figure 15.2.3.2-1 shows the transient void reactivity that would be experienced by the core for various size bubbles. Figures 15.2.3.2-2 and 15.2.3.2-6 show the variation in reactor power for primary scram and secondary scram, respectively. Figures 15.2.3.2-3, -4, -5, -7, -8 and -9 show the maximum fuel, cladding and coolant temperatures for the fuel assemblies with the respective scrams. These maximum temperatures occur at the center of active core for the fuel, at the top of active core for the cladding and at the exit from the upper axial blanket for the coolant temperature. These temperatures occur due to the effect of the reactivity insertion of the bubble

on core power and assumes sodium remains in the subchannels. The additional temperature effect due to the gas bubble blanketing will be discussed later. Figure 15.2.3.2-10, -11 and -12 give maximum fuel, cladding and coolant temperature of the hot pin in the highest power radial blanket assembly for a 4 inch -8 row bubble. As can be seen the results are much less severe in the radial blanket. A 4 inch -8 row bubble is seen to produce fairly insignificant temperature increases in the fuel assembly (less than 35°F increase in cladding temperature). This size bubble (contains over 3 cubic feet of gas at the inlet) should be well beyond the size considered, even on a hypothetical bases, since as indicated in Section 15.2.3.2.1, there is essentially no source for coherent bubbles of even a small fraction of its size. As can be seen by comparing the primary and secondary scram figures, there is very little difference between the results. The reason for this effect is that the reactivity perturbation due to the transient is about over before the control rods begin insertion (reactivity insertion due to a bubble is over in less than 0.3 second). Figure 15.2.3.2-13 gives the maximum cladding temperature increase due to the insulation effect of the bubble passing along the cladding surface. These temperatures were calculated using the transient reactor power changes and the analysis conservatively assumes that all the heat lost from the fuel goes into increasing the cladding temperature. The conservatism is in part based on the fact that some film of sodium will adhere to the clad surface which will provide heat removal during bubble passage. For a 4 inch -8 row bubble, the cladding temperature increase due to the blanketing can be seen to be less than 33°F. The total increase in cladding temperature for this size would be this temperature rise added to the 35°F increase found from the FORE-II analysis (which considers the temperature rise if coolant flows along the entire pin length and the reactor power changes as shown by Figures 15.2.3.2-2 and 15.2.3.2-6).

15.2.3.2.3 Conclusions

As discussed above, there is no identifiable source that would produce large coherent gas bubbles in the CRBR Primary Loops, and even if there were, a large bubble would be dispersed in the inlet plenum. Various size bubbles were analyzed, however, to show their effect on the CRBRP core.

It was found that a bubble even as large as 4 inches -8 rows would cause rather minor consequences to the cladding (a maximum temperature increase of about 68°F). The maximum cladding temperature increases found to result for the various cases are given in Table 15.2.3.2-2. There is little difference between the results for primary and secondary scram since the reactivity perturbation is almost over before the control rods begin insertion. None of the limits for extremely unlikely events in Table 15.1.2-2 would be violated by this transient. A description of how this type of event is incorporated in the pin cladding structural design evaluation is given in Section 4.2.1.3.

Reference

1. J. Muraka, et al., "Gas Bubble Dispersion Test Reactor Inlet Model", HEDL-TME 71-69, May, 1971.

TABLE 15.2.3.2-1

END OF EQUILIBRIUM CYCLE SODIUM VOIDING REACTIVITY $\Delta K/K/Cm^3 \times 10^8$

ΔZ Inches	Row 2	Row 3	Row 4	Row 5	Row 6	Row 7	Row 8	Row 9	Row 10	Row 11	Row 12
3.194	-0.0084	-0.0083	-0.0036	-0.0051	-0.0058	-0.0030	-0.0034	-0.0035	-0.0001	0.0	0.0
3.841	-0.0266	-0.0285	-0.0172	-0.0197	-0.0205	-0.0116	-0.0135	-0.0146	-0.0014	-0.0003	0.0
3.517	-0.0967	-0.1042	-0.0858	-0.0865	-0.0849	-0.0629	-0.0721	-0.0672	-0.0082	-0.0016	-0.0003
2.518	-0.2819	-0.2925	-0.3700	-0.3003	-0.2762	-0.2796	-0.3089	-0.2562	-0.0369	-0.0057	-0.0011
2.568	-0.2637	-0.3095	-0.2890	-0.3401	-0.3410	-0.3311	-0.4825	-0.5287	-0.0998	-0.0154	-0.0034
2.568	0.2309	0.1897	0.1999	0.1194	0.0548	-0.0172	-0.2524	-0.5183	-0.1349	-0.0212	-0.0045
3.543	0.8927	0.8637	0.7882	0.7205	0.6049	0.4255	0.0579	-0.4611	-0.1641	-0.0274	-0.0063
2.793	1.5913	1.5649	1.4217	1.3458	1.1828	0.8914	0.3741	-0.3684	-0.1787	-0.0086	-0.0085
2.793	2.0440	2.0138	1.8435	1.7541	1.5595	1.2038	0.5754	-0.3192	-0.1854	-0.0338	-0.0093
2.793	2.3387	2.3077	2.1181	2.0184	1.8023	1.4071	0.7063	-0.2902	-0.1905	-0.0343	-0.0096
1.862	2.3511	2.4000	2.2211	2.1162	1.8911	1.4817	0.7533	-0.2803	-0.1921	-0.0342	-0.0096
2.793	2.4232	2.3600	2.1914	2.0846	1.8596	1.4550	0.7338	-0.2838	-0.1903	-0.0339	-0.0095
2.793	2.2253	2.1400	2.0013	1.8962	1.6819	1.3049	0.6331	-0.3034	-0.1839	-0.0329	-0.0092
2.793	1.8610	1.6800	1.6570	1.5586	1.3656	1.0384	0.4572	-0.3380	-0.1735	-0.0312	-0.0085
3.624	1.2382	1.2019	1.0862	1.0031	0.8487	0.6130	0.1815	-0.3974	-0.1552	-0.0276	-0.0071
2.718	0.5959	0.5603	0.5048	0.4283	0.3181	0.1766	-0.1189	-0.4488	-0.1283	-0.0227	-0.0056
2.718	0.5812	0.0526	0.0620	0.0221	-0.0880	-0.1455	-0.1673	-0.4736	-0.0996	-0.0162	-0.0049
4.683	-0.590	-0.0753	-0.0676	-0.1047	-0.1291	-0.1546	-0.2392	-0.2350	-0.0349	-0.0065	-0.0015
4.683	-0.0596	-0.0585	-0.0579	-0.8608	-0.0605	-0.0670	-0.0740	-0.0557	-0.0070	-0.0016	-0.0004
4.683	-0.0621	-0.0576	-0.0544	-0.0508	-0.0436	-0.0377	-0.0299	-0.0176	-0.0024	-0.0006	-0.0002

Note: ΔZ increments start at bottom of lower axial blanket positions and go to top of upper axial blanket.

TABLE 15.2.3.2-2

MAXIMUM CLADDING TEMPERATURE INCREASE DUE TO
VARIOUS SIZE LARGE BUBBLES PASSING THRU CORE

Height of Assembly Row		Max. Cladding Temp. Increase at Hot Spot, (°F)	Volume at Inlet (Ft^3)
4 inches	8	68	3.1
8 inches	4	65	1.1
4 inches	4	35	0.6
1 inch	8	14	0.8

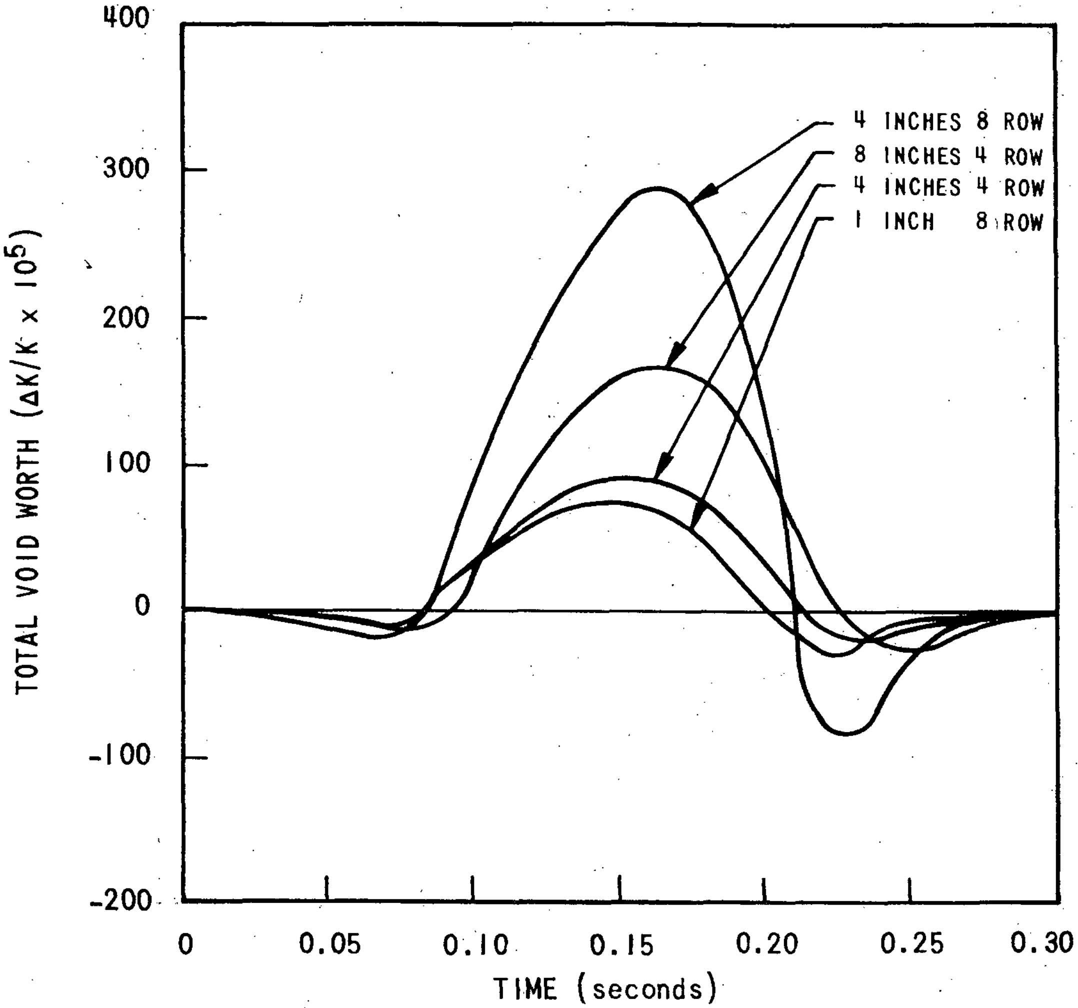

Figure 15.2.3.2-1. Transient Void Reactivity Worth for Various Size Large Bubbles

6727-32

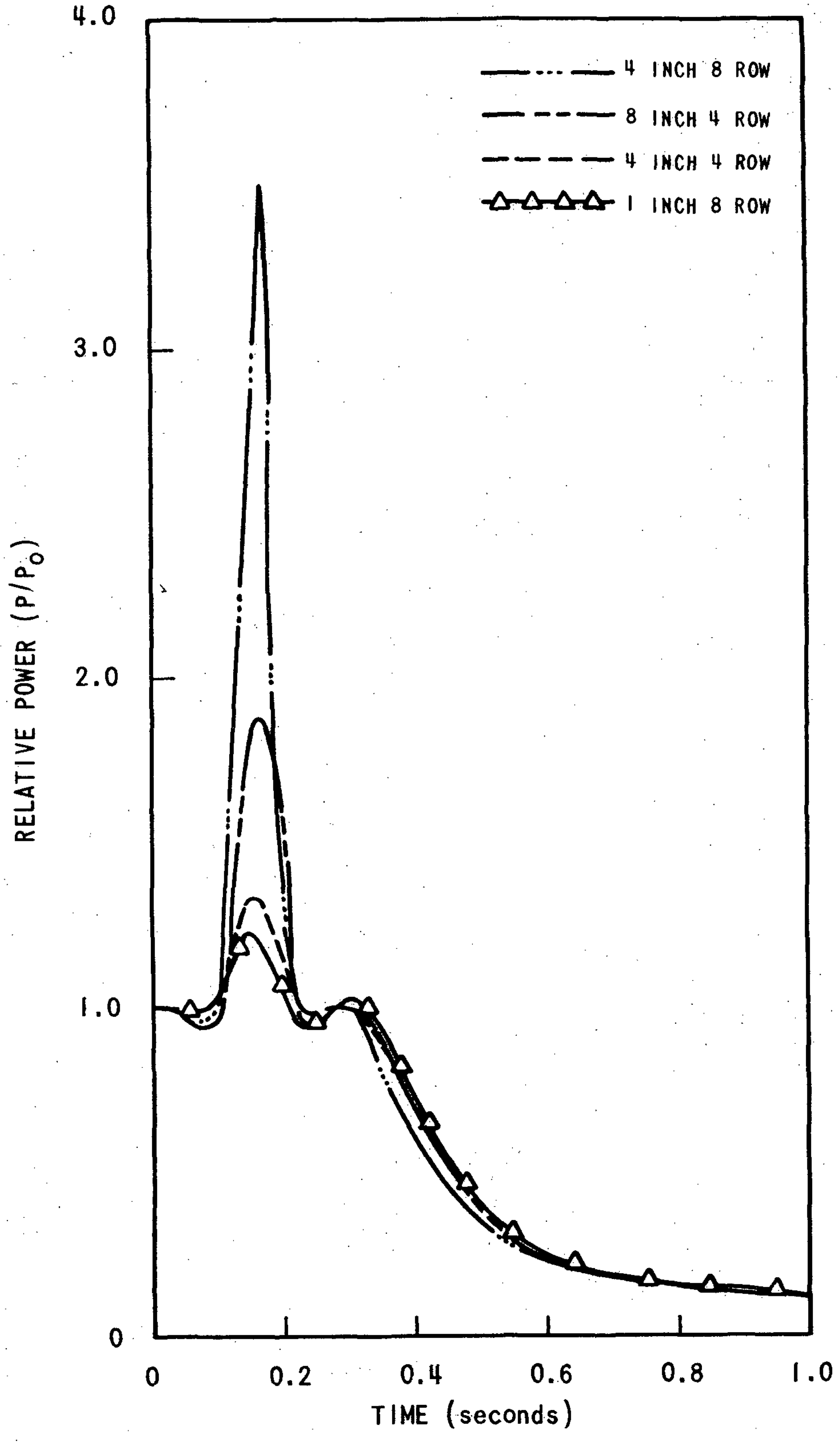

Figure 15.2.3.2-2. Transient Variation in Reactor Power for Various Size Bubbles with Primary Shutdown

6727-33

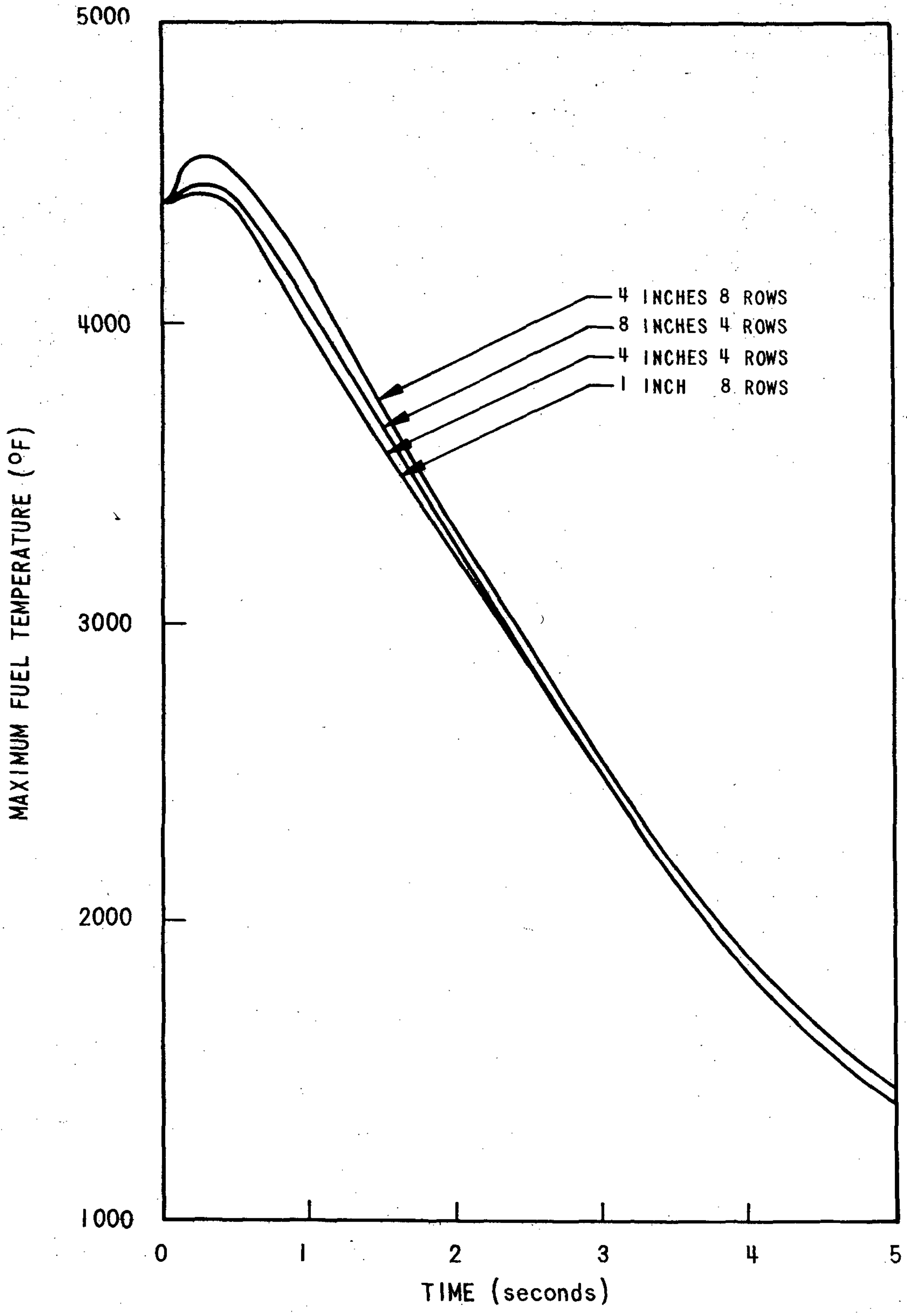

Figure 15.2.3.2-3. Transient Variation in F/A Hot Pin Maximum Fuel Temperature for Various Size Large Bubbles with Primary Scram

6727-34

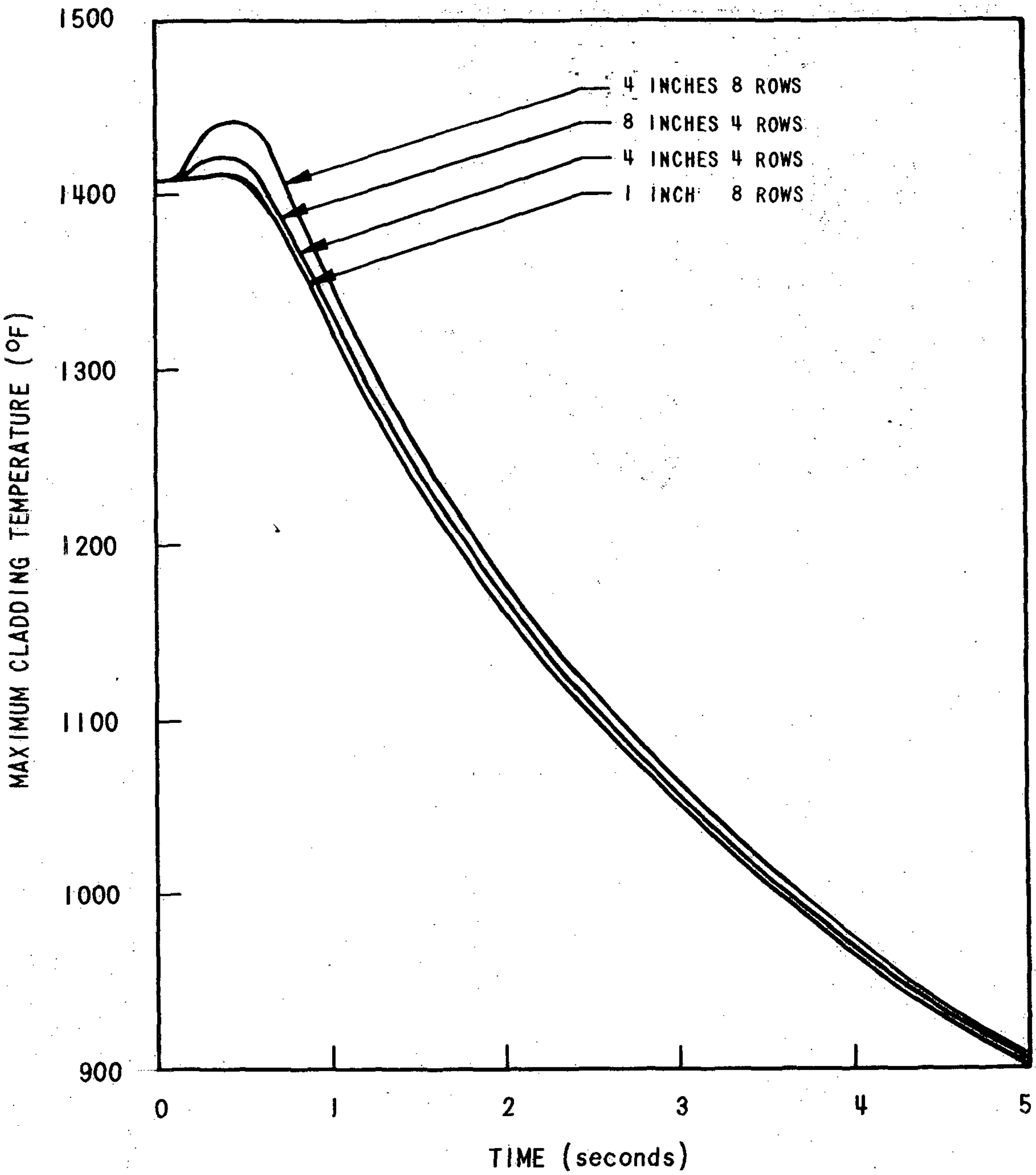

Figure 15.2.3.2-4. Transient Variation in F/A Hot Pin Maximum Cladding Temperature for Various Size Large Bubble with Primary Scram

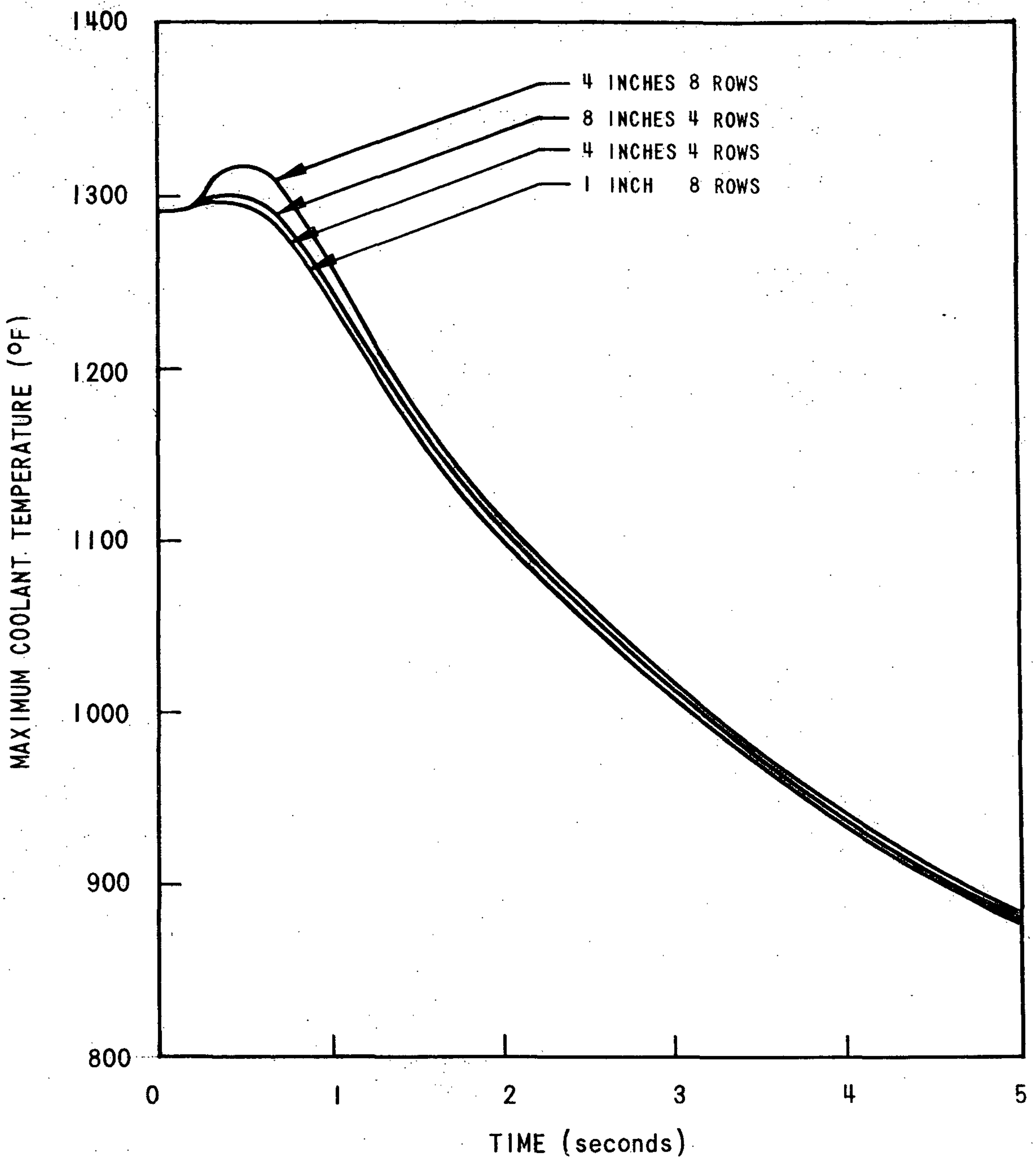

Figure 15.2.3.2-5. Transient Variation in F/A Hot Pin Maximum Coolant Temperature for Various Size Large Bubbles with Primary Scram

6727-36

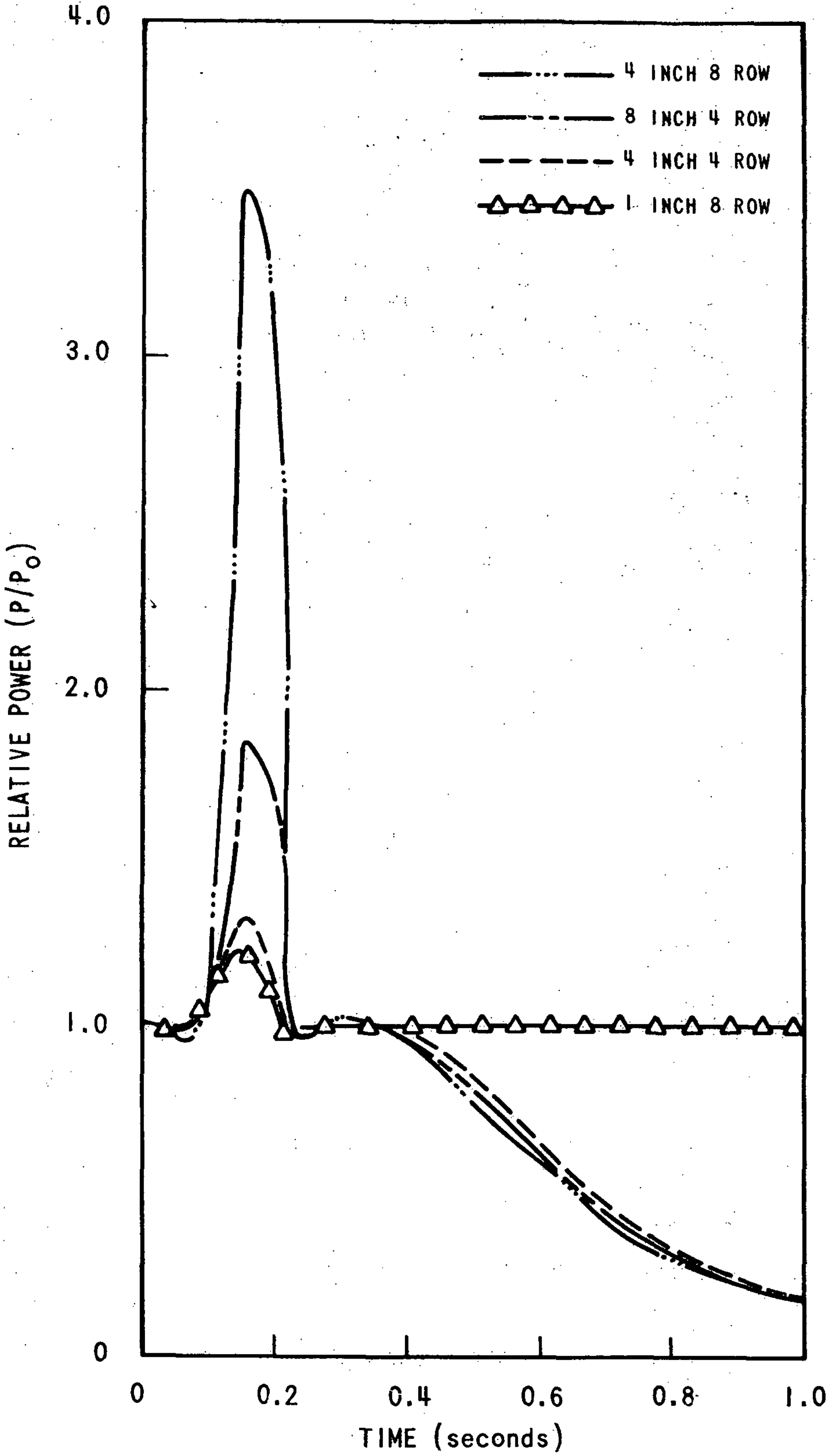

Figure 15.2.3.2-6. Transient Variation in Reactor Power for Various Size Large Bubbles with Secondary Scram

6727-37

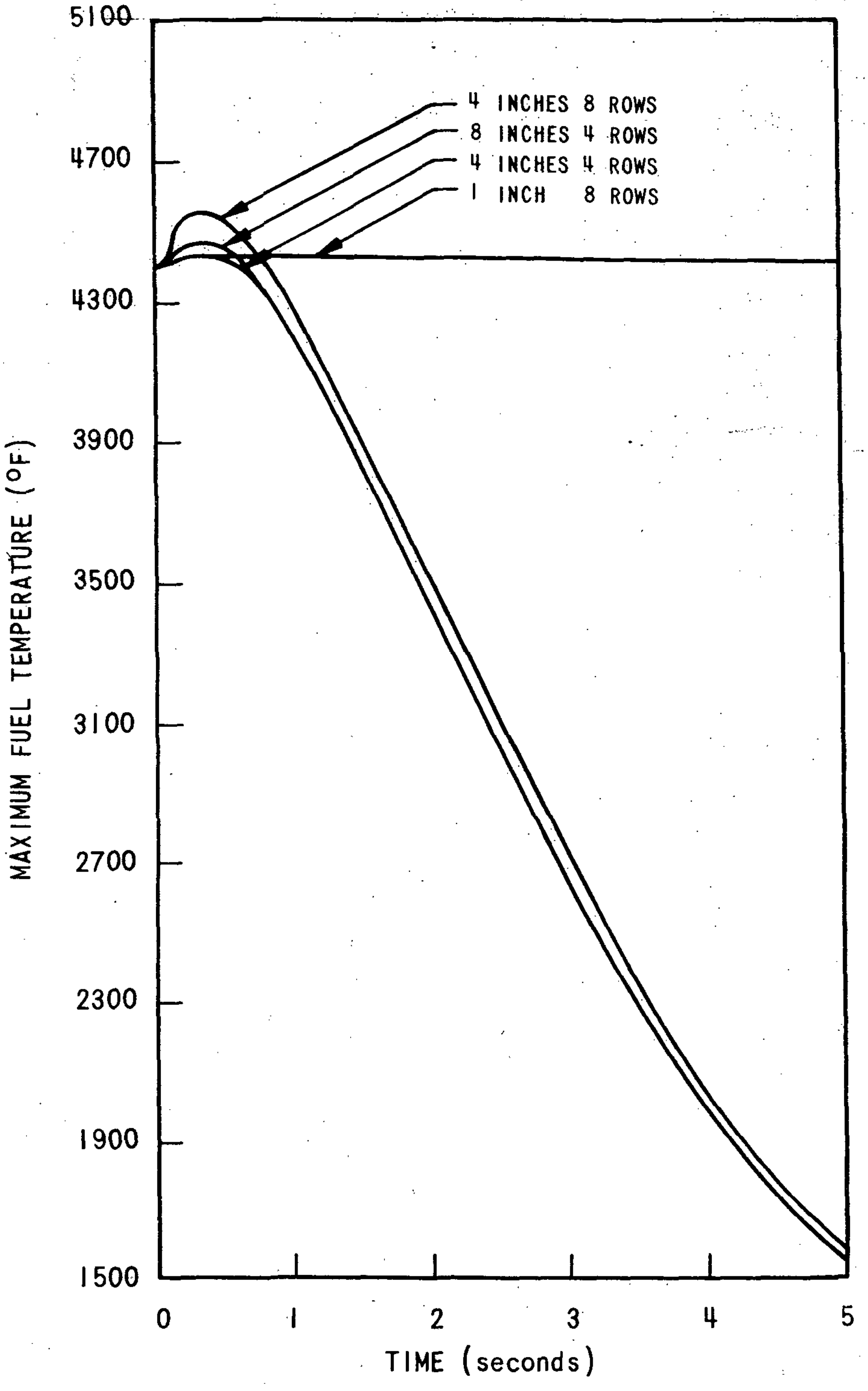

Figure 15.2.3.2-7. Transient Variation in F/A Hot Pin Maximum Fuel Temperature for Various Size Large Bubbles with Secondary Scram

6727-38

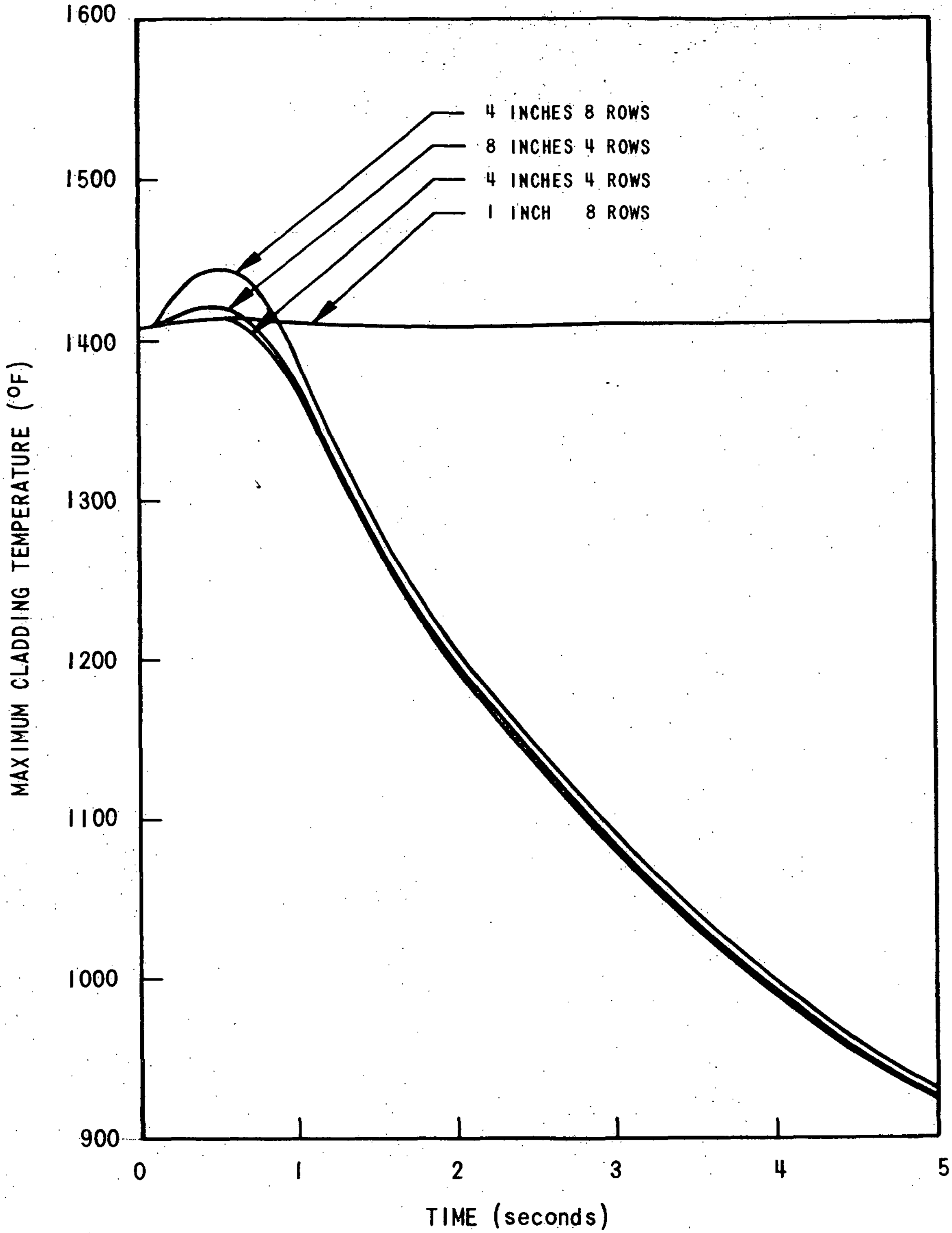

Figure 15.2.3.2-8. Transient Variation in F/A Hot Pin Maximum Cladding Temperature for Various Size Large Bubbles with Secondary Scram

6727-39

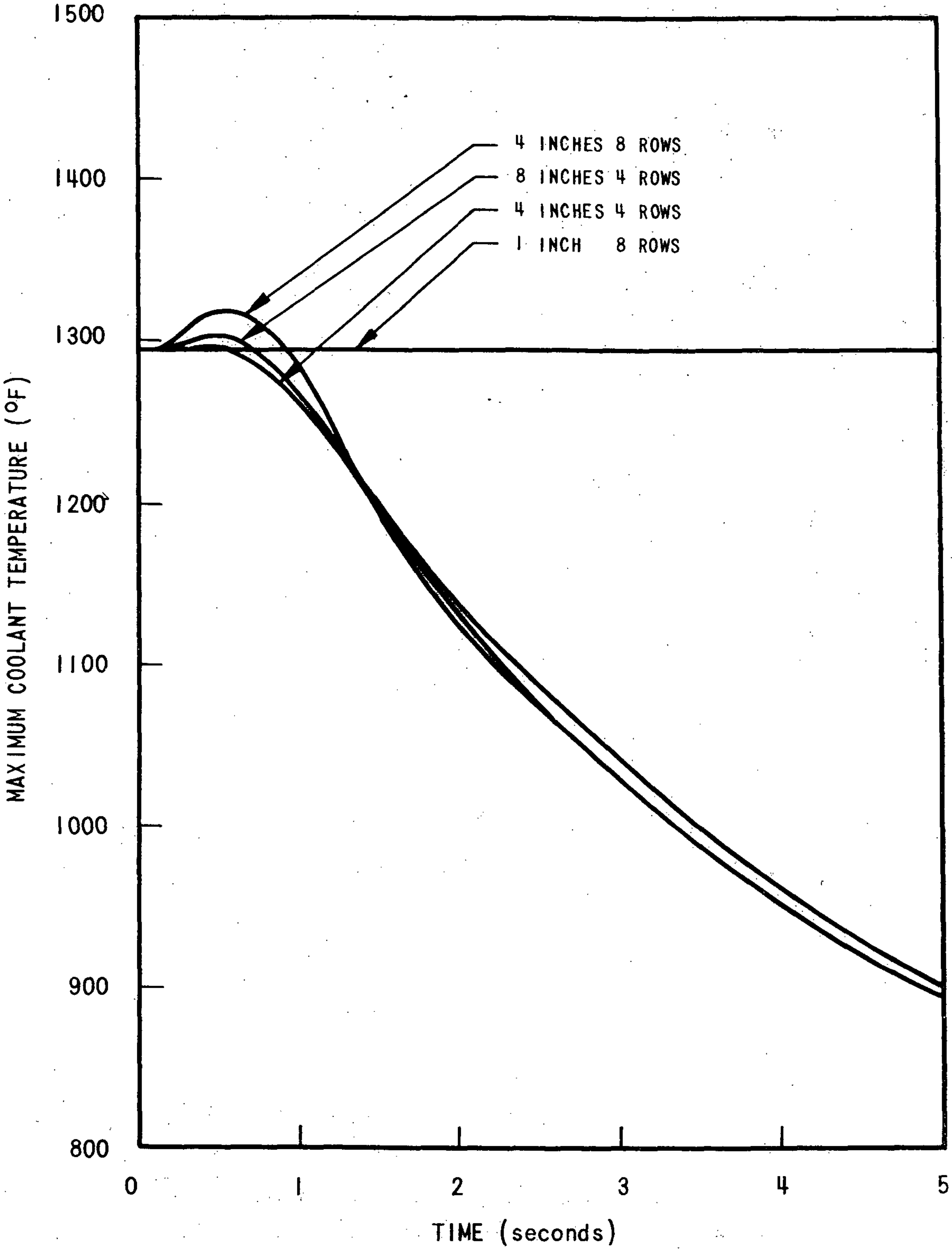

Figure 15.2.3.2-9. Transient Variation in F/A Hot Pin Maximum Coolant Temperature for Various Size Large Bubbles with Secondary Shutdown

6727-40

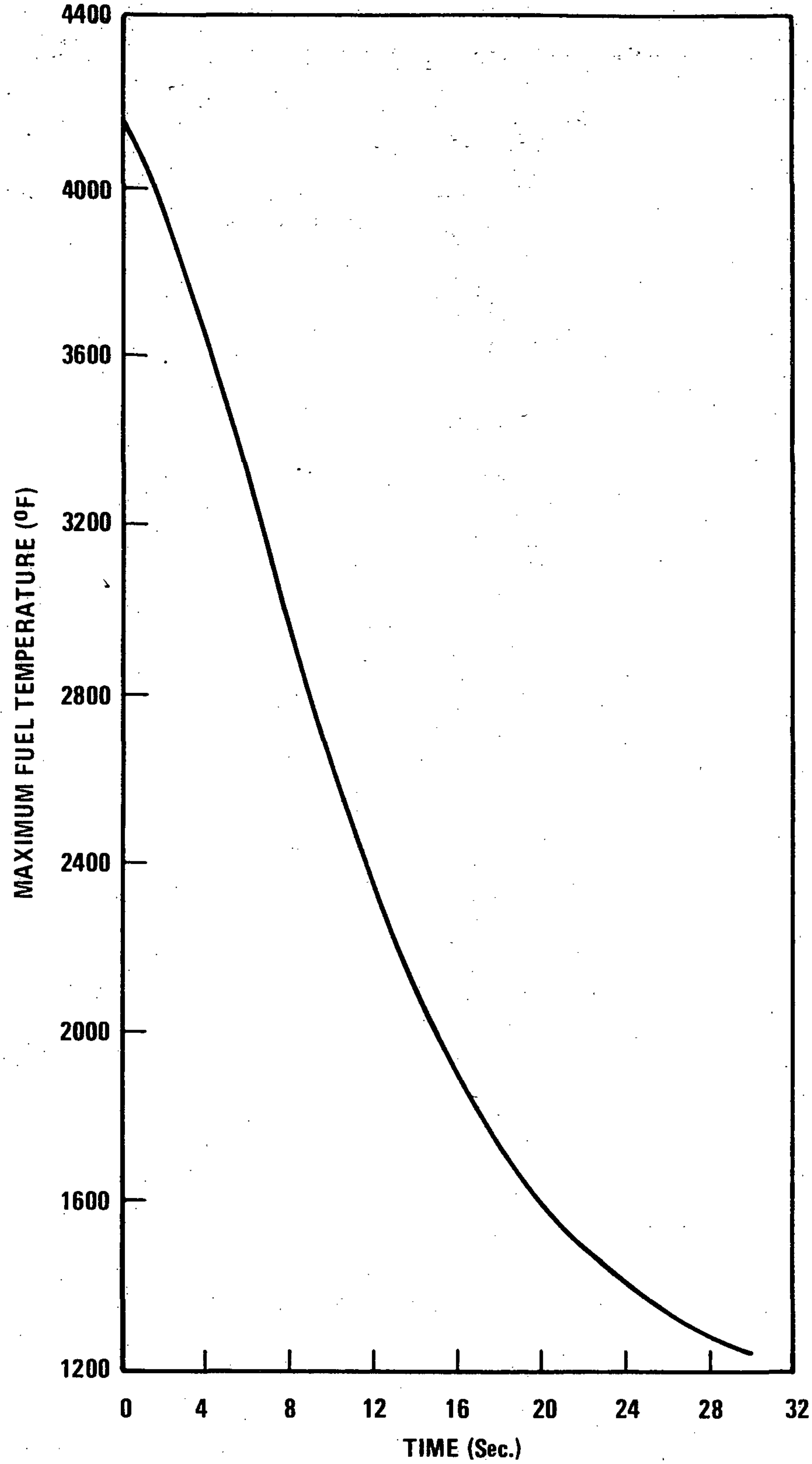

Figure 15.2.3.2-10. Transient Variation in RB/A Hot Pin Maximum Fuel Temperature for 4 Inch - 8 Row Bubble With Secondary Scram

6727-67

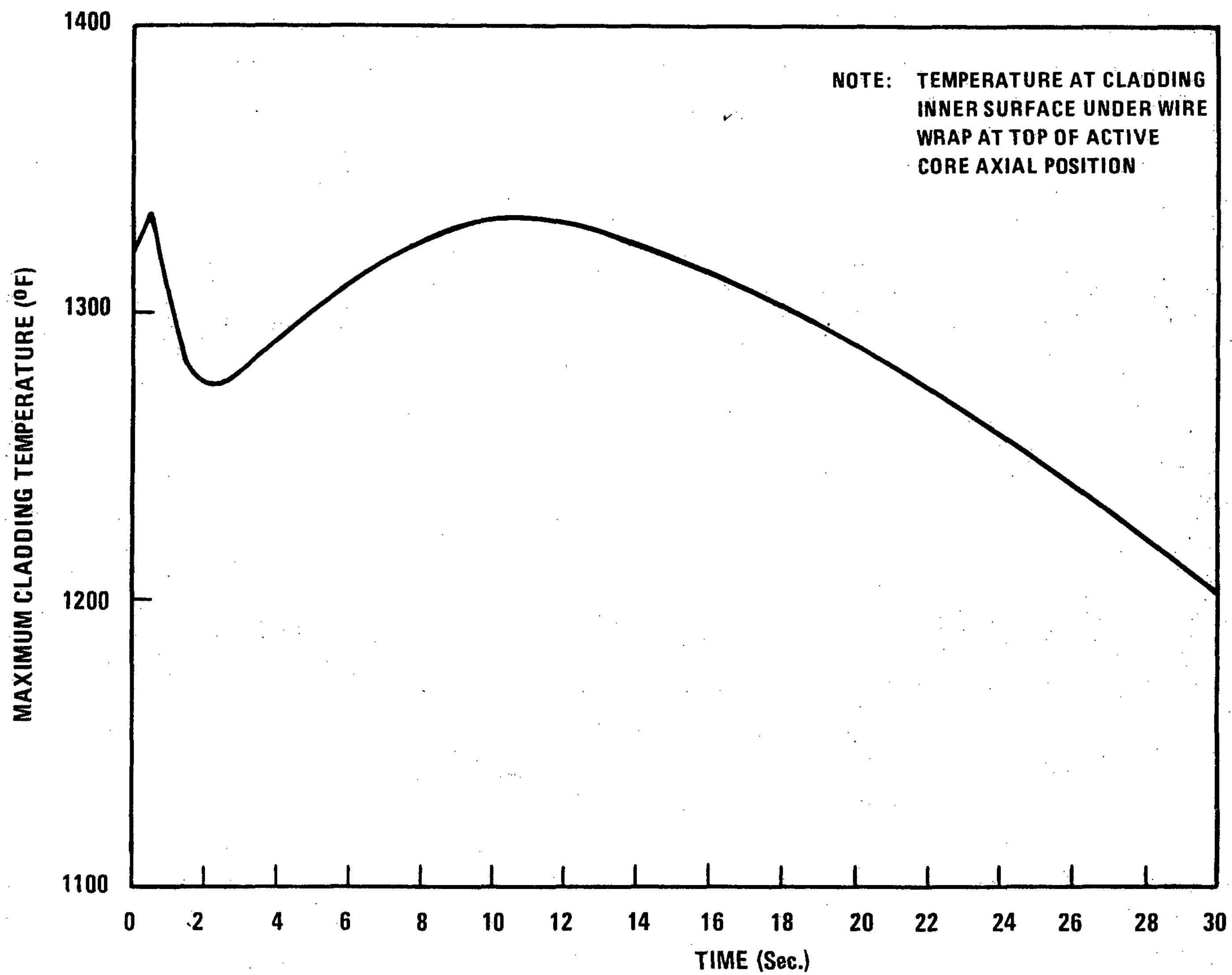

Figure 15.2.3.2-11. Transient Variation in RB/A Maximum Cladding Temperature for 4 Inch - 8 Row Bubble With Secondary Scram

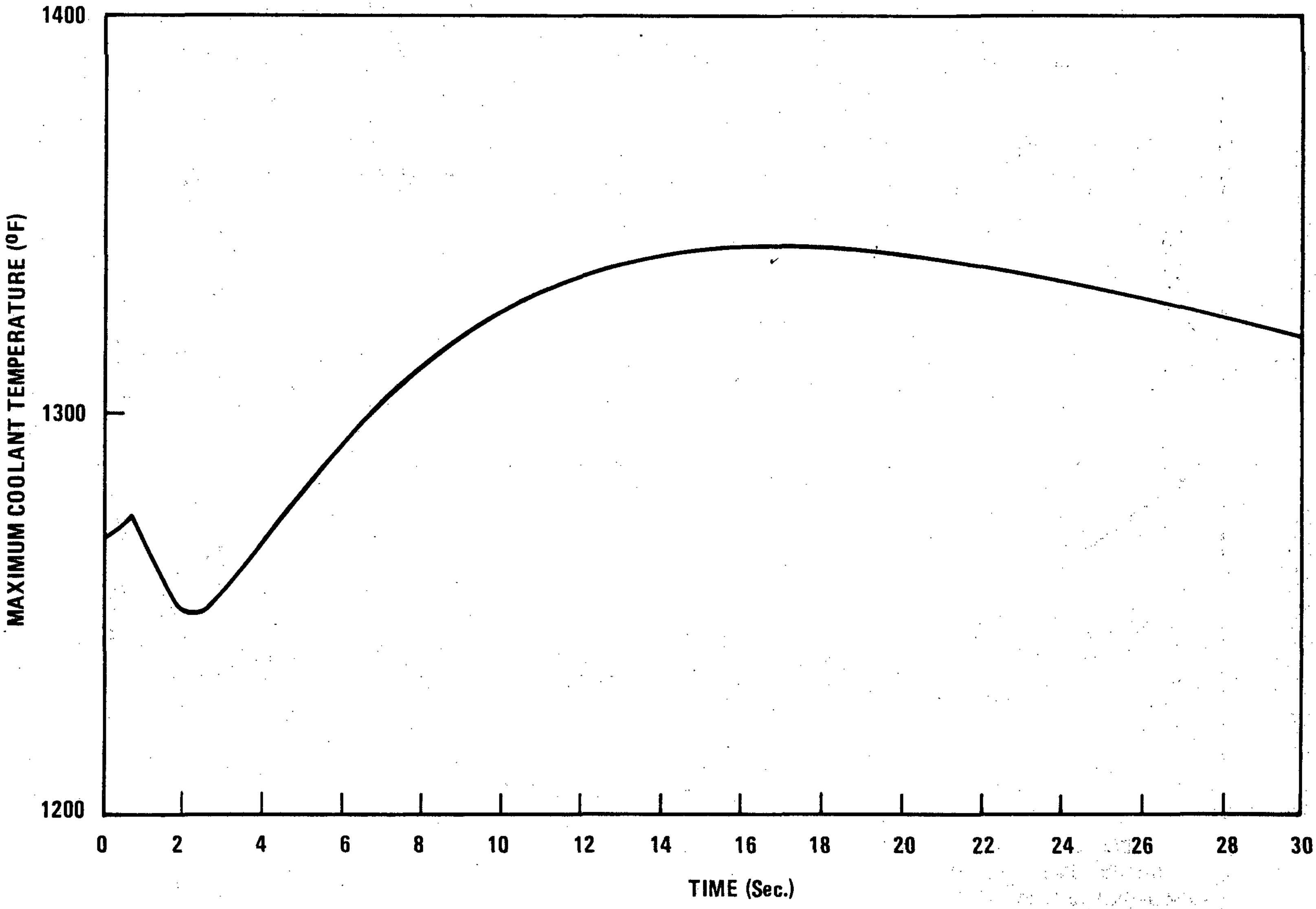

Figure 15.2.3.2-12. Transient Variation in RB/A Hot Pin Maximum Coolant Temperature for 4 Inch - 8 Row Bubble With Secondary Scram

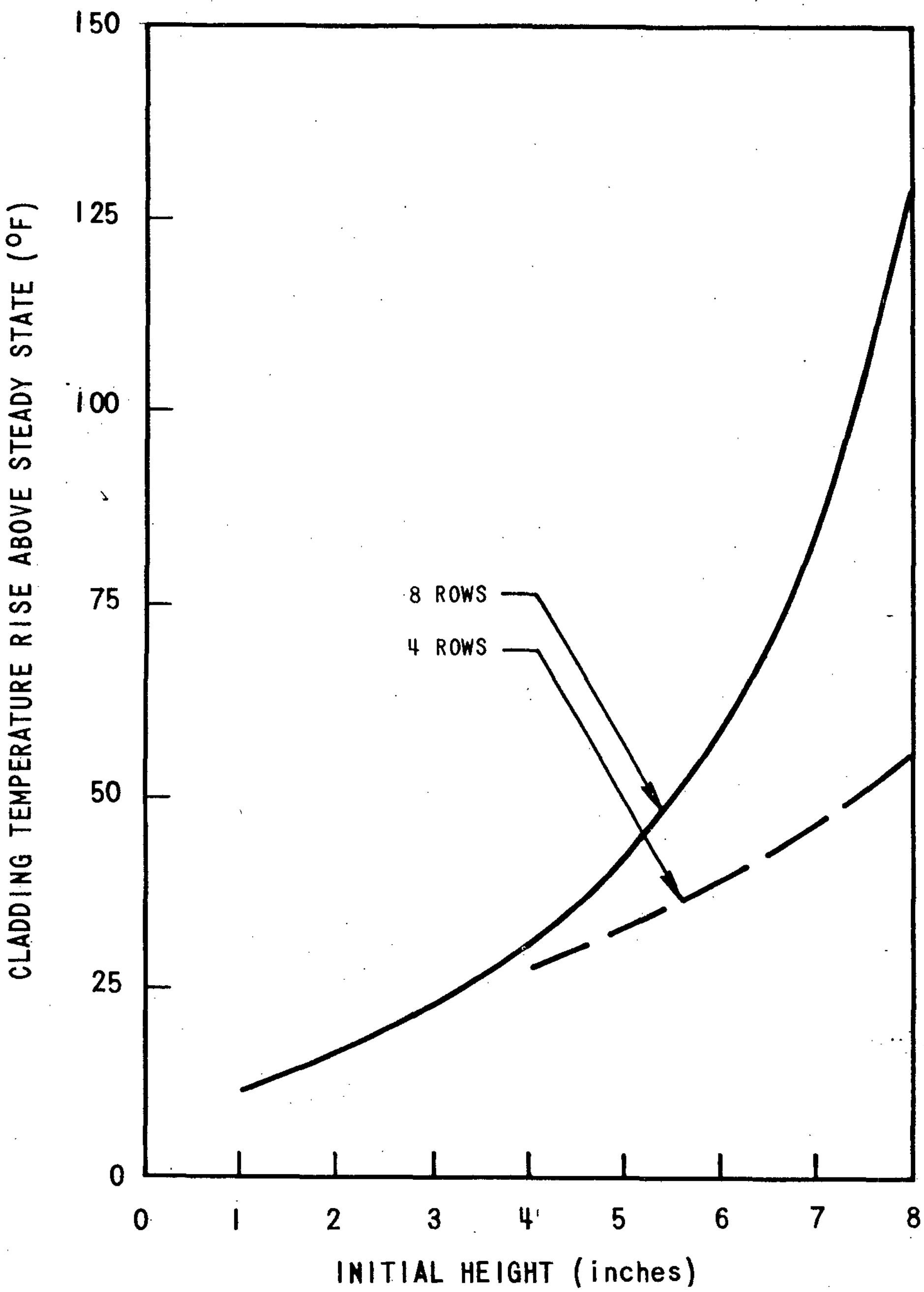

Figure 15.2.3.2-13. Maximum Cladding Temperature Increase Due to Bubble Insulation Effect at Top of Active Core

6727-44

15.2.3.3 Seismic Reactivity Insertion (Fuel, Radial Blanket and Control Rod Assemblies) - SSE

15.2.3.3.1 Identification of Causes and Accident Description

For the Safe Shutdown Earthquake (SSE) several conditions exist that compound the severity of the event. First, the earthquake can produce a loss of off-site electrical power causing a loss of power to the pumps and consequently a decay of the primary coolant flow. Second, the acceleration forces of the earthquake can cause compaction of the core due to closing of radial gaps between the assemblies at the above core load pad (ACLP) position. This can result in a net positive step reactivity insertion to the core. Third, when the control rods are scrammed, the rate of inward motion is decreased from the normal rate due to a retarding force resulting from seismic induced impacts of the control rod assembly duct and driveline on surrounding guide structures.

The first two subsystems initiate automatic reactor scram (the operator can also initiate scram). Once power to the pumps is lost (it is assumed that the pumps remain operable following an SSE) a loss of electrical power (LOEP) trip system initiates scram after a 0.5 second delay. If a step reactivity insertion occurs the primary rods are scrammed when the reactor reaches 115% of full power.

The worst combination of the above events with respect to core temperatures during the SSE would be to assume that a step reactivity insertion occurs 0.5 second after the power to the pumps is lost. If the step occurs earlier than this the core flow will be at a higher level when the control rod insertion begins and the resultant temperatures for the event would be lower. This is due to the fact that for steps on the order of 30¢ the power rises very rapidly (in less than 0.1 second) and thus scram would be initiated by the 15% overpower condition almost instantaneously. If the step occurs later then 0.5 second the signal due to LOEP would have already started the plant protection system scram and the reactor power would be dropping below its initial value at the time of the reactivity insertion and again less severe temperature conditions would result.

15.2.3.3.2 Analysis of Effects and Consequences

Since the highest power fuel assembly occurs at the beginning-of-equilibrium cycle (BOEC), the transients were analyzed for this particular worst period in core life. A minimum value Doppler coefficient of -0.005 was used for this core condition. This value is obtained by decreasing the nominal Doppler coefficient by 20% for uncertainties as discussed in Section 4.3.2.3. Scram was taken at 15% overpower. [Note: For the worst case chosen both the 15% overpower trip and the LOEP trip would cause the control rod insertion at nearly the same time as discussed in Section 15.2.3.3.1.] The maximum worth control assembly was assumed to be stuck and the shutdown worth was decreased by the appropriate amount.

Figure 15.2.3.3-1 shows the normal insertion rate and the seismic insertion rate for the primary rods. These insertion rates were calculated with the CRAB code (described in Section 4.4.3) using the seismic force-time history curves in Section 3.9.3. As can be seen, at the BOEC the primary rods are banked such that some are 11", 23" and 36" withdrawn while at full power. Also, it can be noted that the earthquake causes only about a 0.32 second increase in scram time for an initially fully withdrawn control rod to be inserted its full 36".

An analysis of various size step insertions occurring during the SSE has been performed with FORE-II (Ref. 1) (see Appendix A). The results of a parametric range of step insertions up to 90¢ are shown by Figure 15.2.3.3-2. Figures 15.2.3.3-3 to -6 show the variation in reactor power and fuel assembly hot pin maximum fuel, cladding and coolant temperatures (with 3σ hot channel factors).

The size step insertion that would be expected to occur during the SSE has not been determined yet since it is a complicated study involving such effects as the thermal expansion of all the assemblies at their discrete temperatures, swelling of the assemblies, manufacturing tolerances and gap restrictions imposed by the core restraint system of the assemblies. This information goes into determining assembly bowing profiles and interassembly loads and gaps due to the earthquake forces. It can be seen from Figure 15.2.3.3-2 that, even if a step insertion as large as 60¢* should occur, the maximum cladding temperature would be less than 1505°F. The duration of the cladding temperature above its initial steady state value for this case would be about 1.0 second.

Several runs for the SSE were also made for the highest power radial blanket assembly. For a 60¢ step, the hot pin maximum cladding temperature was less than 1370°F. The radial blanket hot pin maximum temperatures for 60 and 90¢ steps are given by Figures 15.2.3.3-7, -8 and -9 for the fuel, cladding and coolant. The second temperature peak on Figure 15.2.3.3-8 is due to the slow release of internally stored heat in the radial blanket pins (i.e., radial blanket pins have a 0.52" O.D. as compared to 0.23" for fuel pins) relative to the flow decay when the primary pumps are tripped.

61| As indicated in Section 15.1, parametric studies were performed to show the effect of using "minimum required" primary control rod shutdown rate values instead of the "expected" values (both having the highest worth rod assumed to be stuck). The temperatures described thus far in this section have been based on the expected rates of shutdown worth which, as described in this earlier section, are felt to give the more realistic evaluation of the transient. Figure 15.2.3.3-10 shows the fuel assembly hot spot cladding temperature for the two cases. As can be seen, for a 60¢ step the maximum temperature would increase from about 1505°F to 1570°F. As described in Section 15.1.2, the most limiting cladding temperature requirement for an extremely unlikely event is that no sodium boiling is allowed to occur. In order to achieve sodium saturation temperature during the SSE the cladding temperature must be in excess of 1700°F.

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*The value calculated for FFTF under these conditions was 35¢.

15.2.3.3.3 Conclusions

A conservative analysis was performed for the extremely unlikely event of an SSE considering the compound effects of core flow decay due to loss of power to the pumps, step reactivity insertion due to changes in core configuration and the decrease in the control rod insertion rate.

It was found that, even if a 60¢ step reactivity insertion occurred for the SSE, the fuel assembly maximum cladding temperature would not exceed 1505°F using expected primary control rod shutdown rates. Although an allowable consequence of an extremely unlikely event such as the SSE is cladding failure up to loss of in-place coolable geometry, no significant degradation of the cladding or failure would be expected for a short term temperature rise of this magnitude. In the long term it is assumed that the pumps remain operable following an SSE and adequate cooling is provided to the core. A description of how this type event is incorporated in the pin cladding structural design evaluation is given in Section 4.2.1.3.

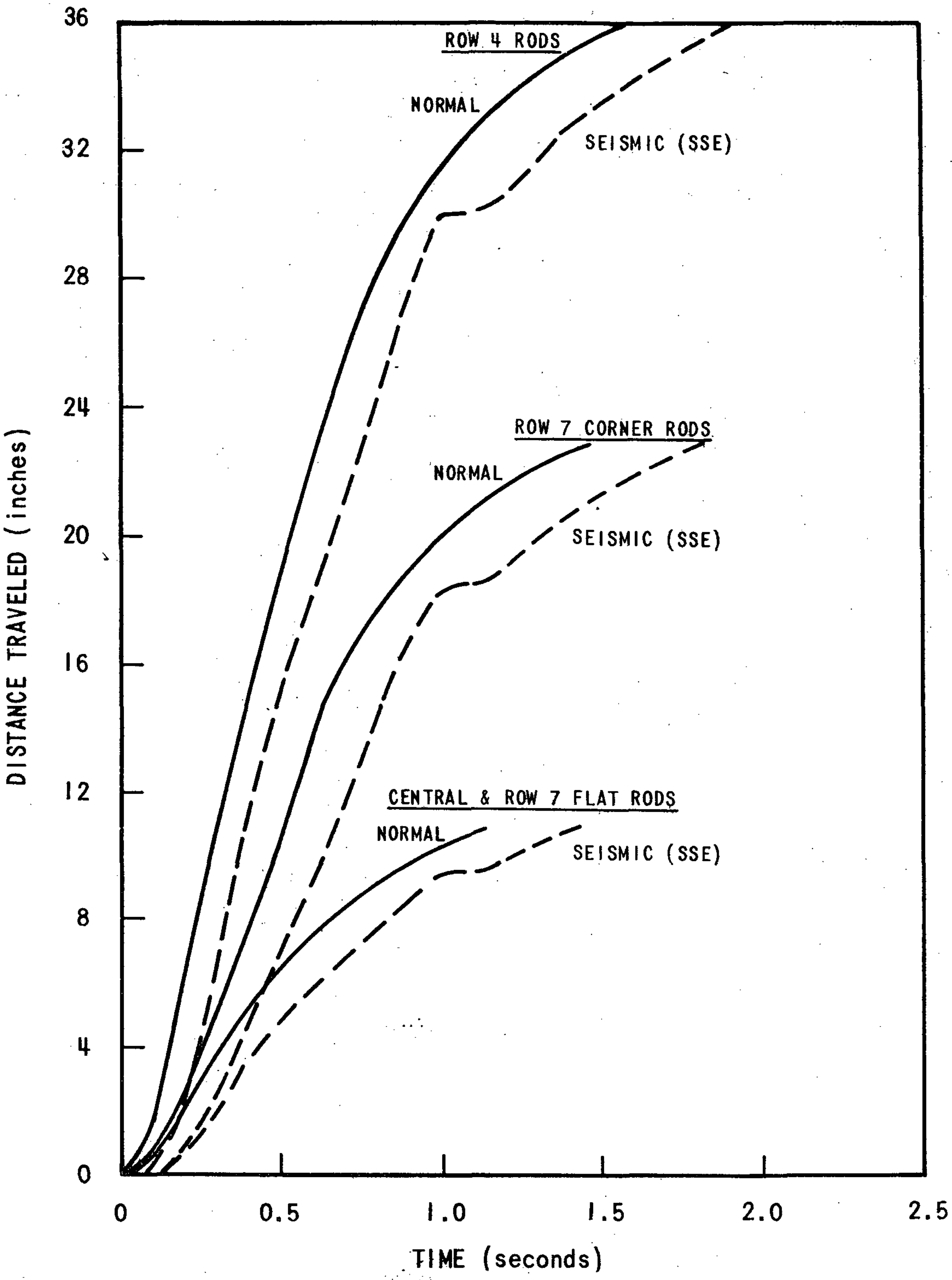

Figure 15.2.3.3-1. Insertion of Primary Control Rods Under SSE Conditions Compared to Normal Insertion

6727-45

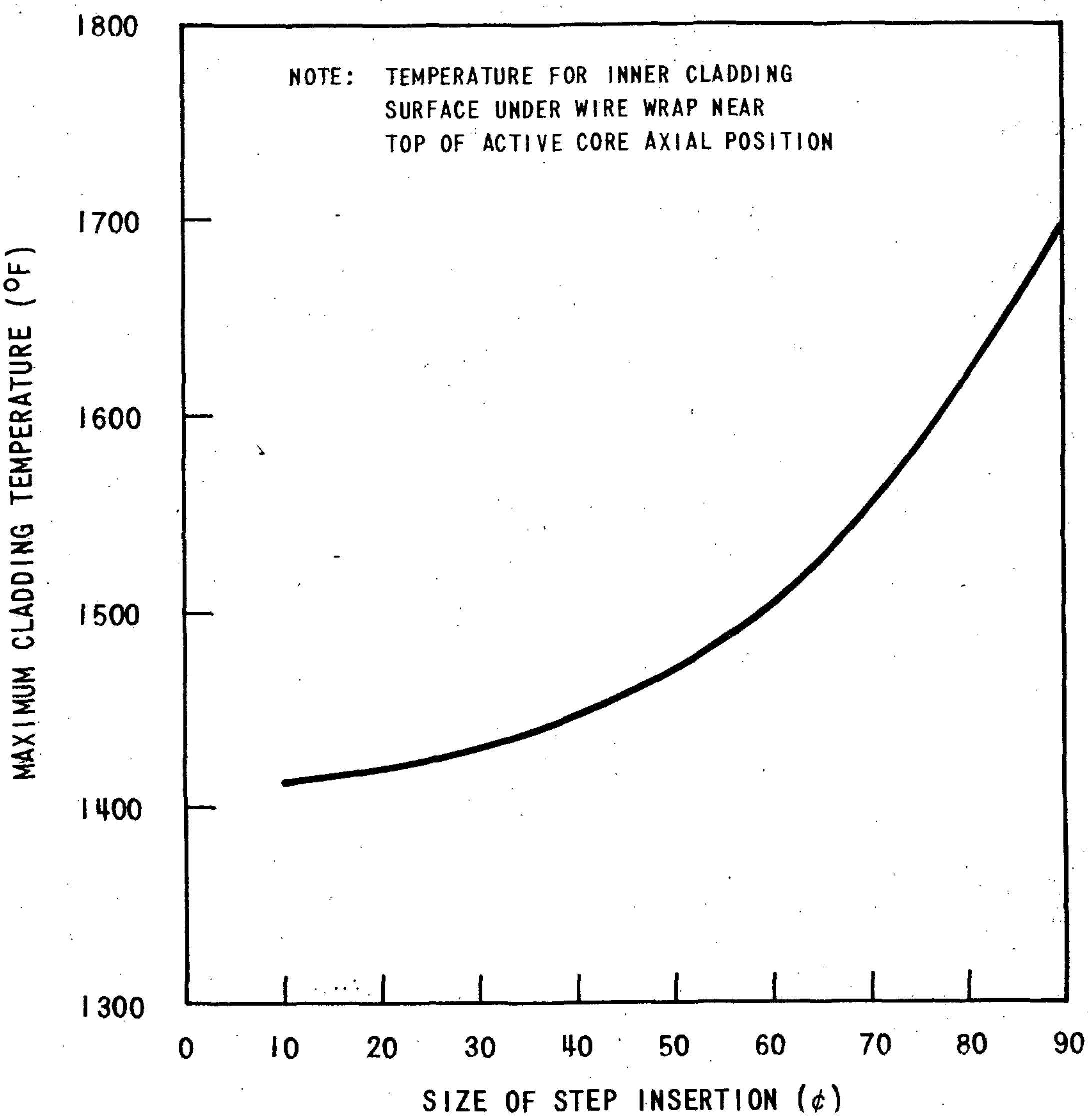

Figure 15.2.3.3-2. Fuel Assembly Maximum Cladding Temperature as a Function of Size of Step Reactivity Insertion

6727-46

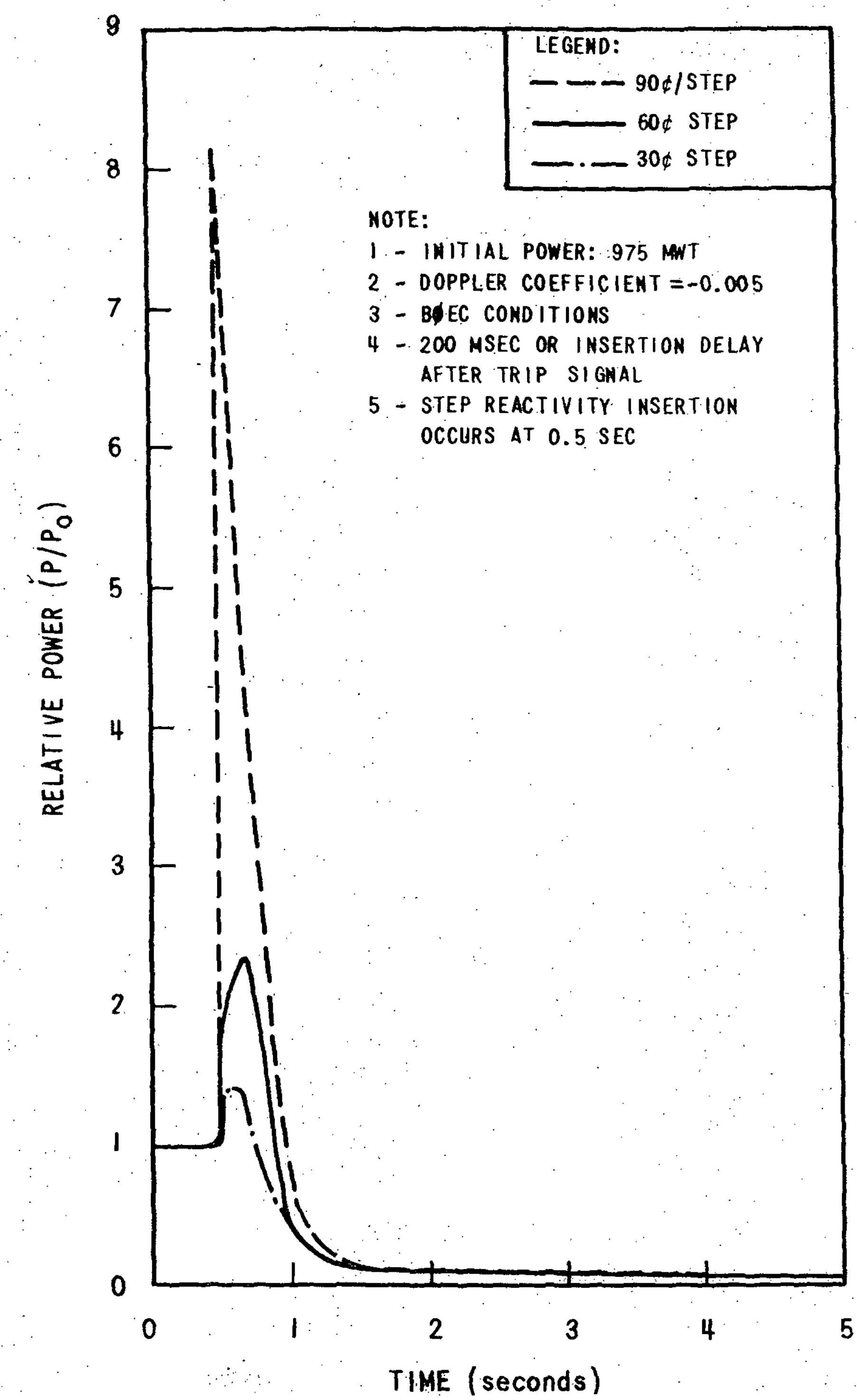

Figure 15.2.3.3-3. Variation in Reactor Power for SSE Transient

6727.47

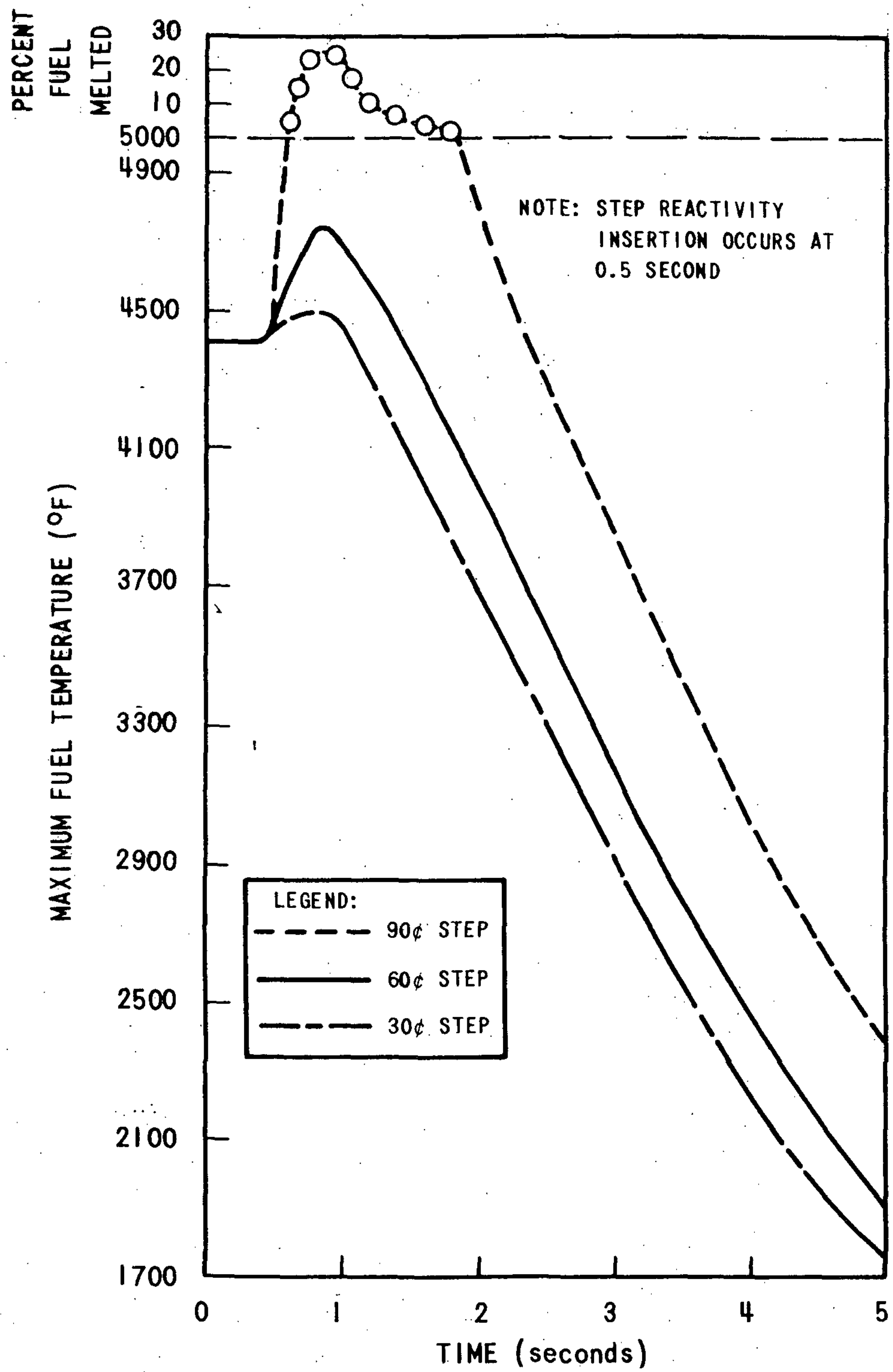

Figure 15.2.3.3-4. Variation in F/A Hot Channel Maximum Fuel Temperature for SSE Transient

6727-48

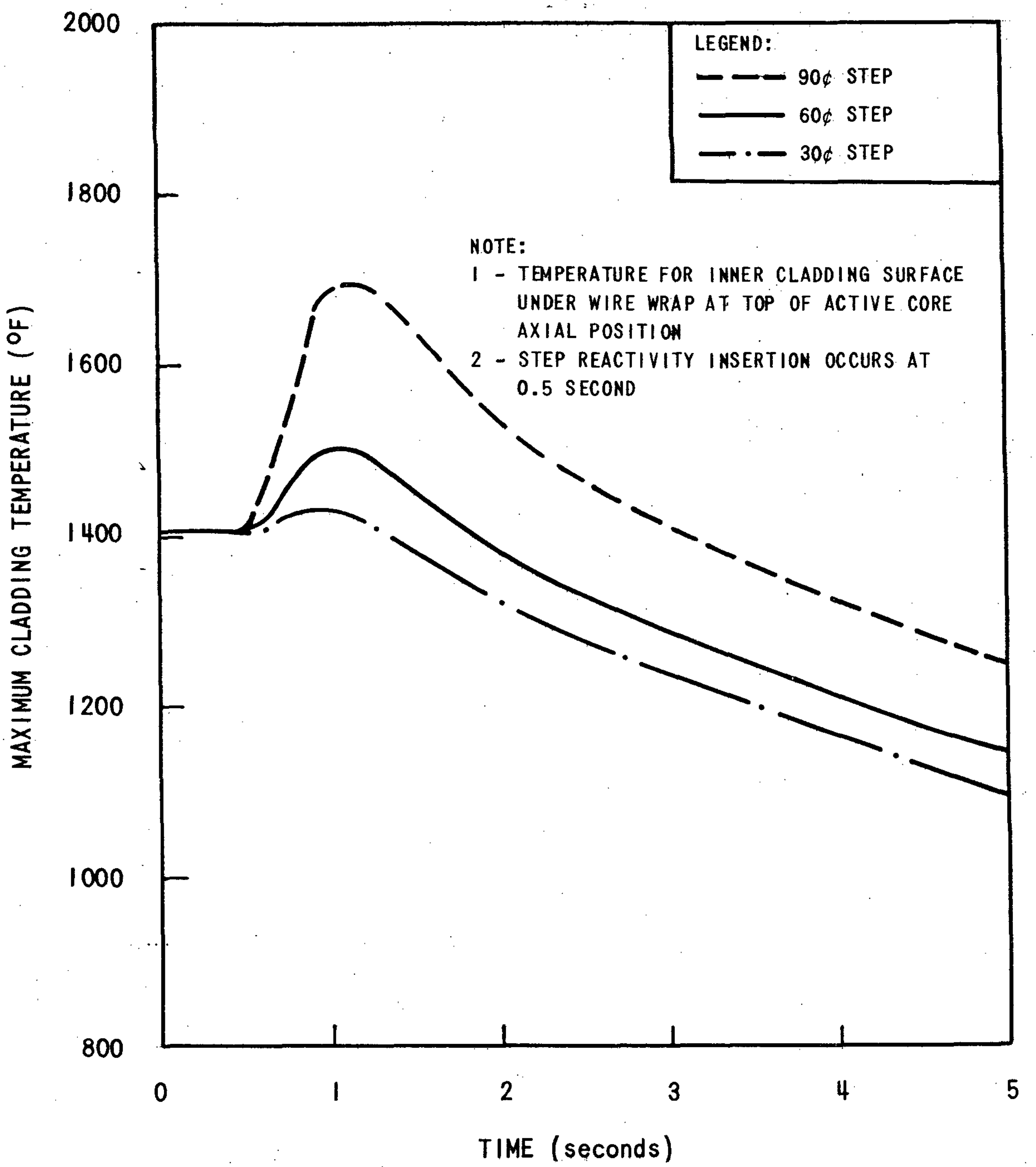

Figure 15.2.3.3-5. Variation in F/A Hot Channel Maximum Cladding Temperature for SSE Transient

6727-49

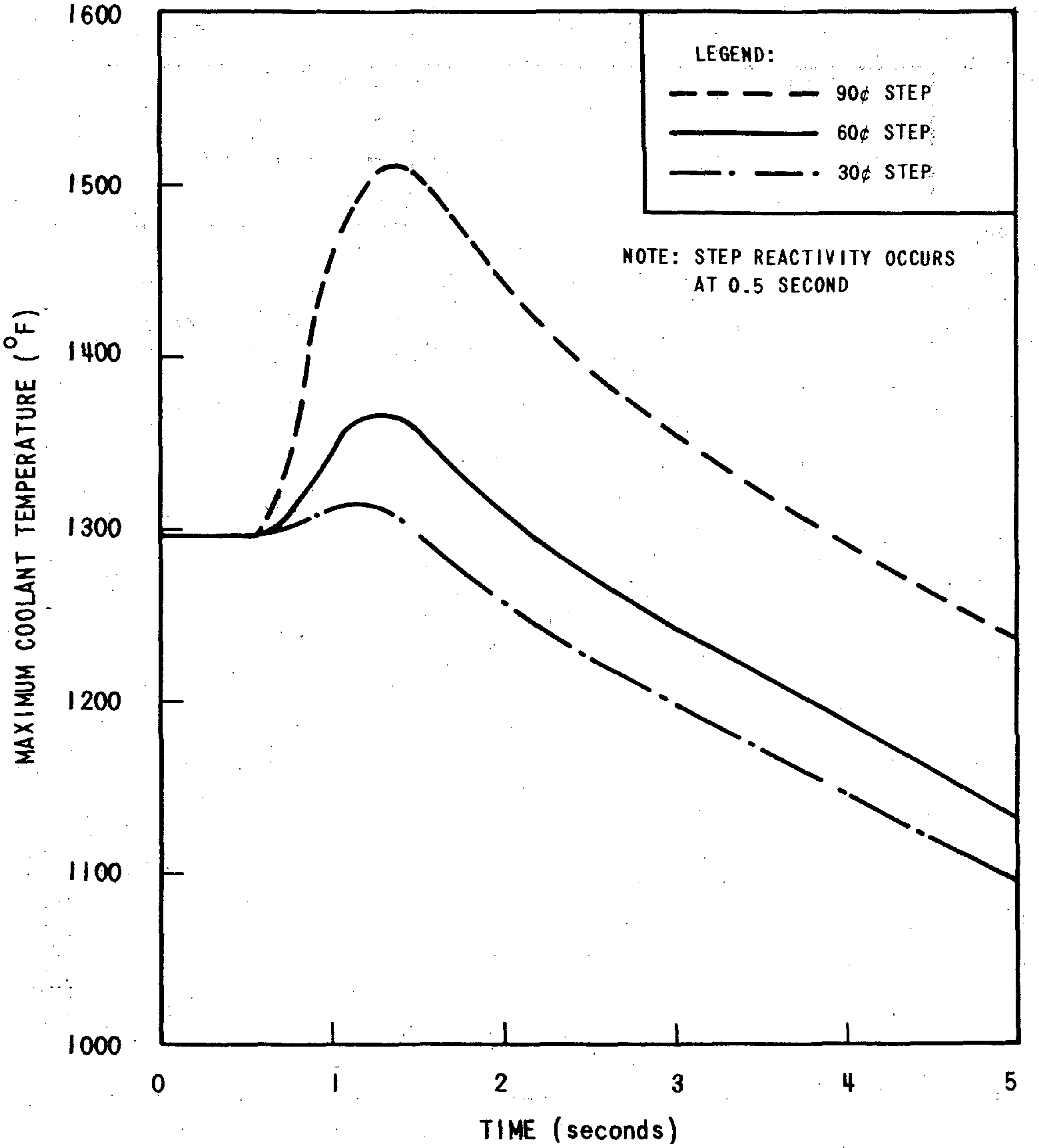

Figure 15.2.3.3-6. Variation in F/A Hot Channel Maximum Coolant Temperature for SSE Transient

6727-50

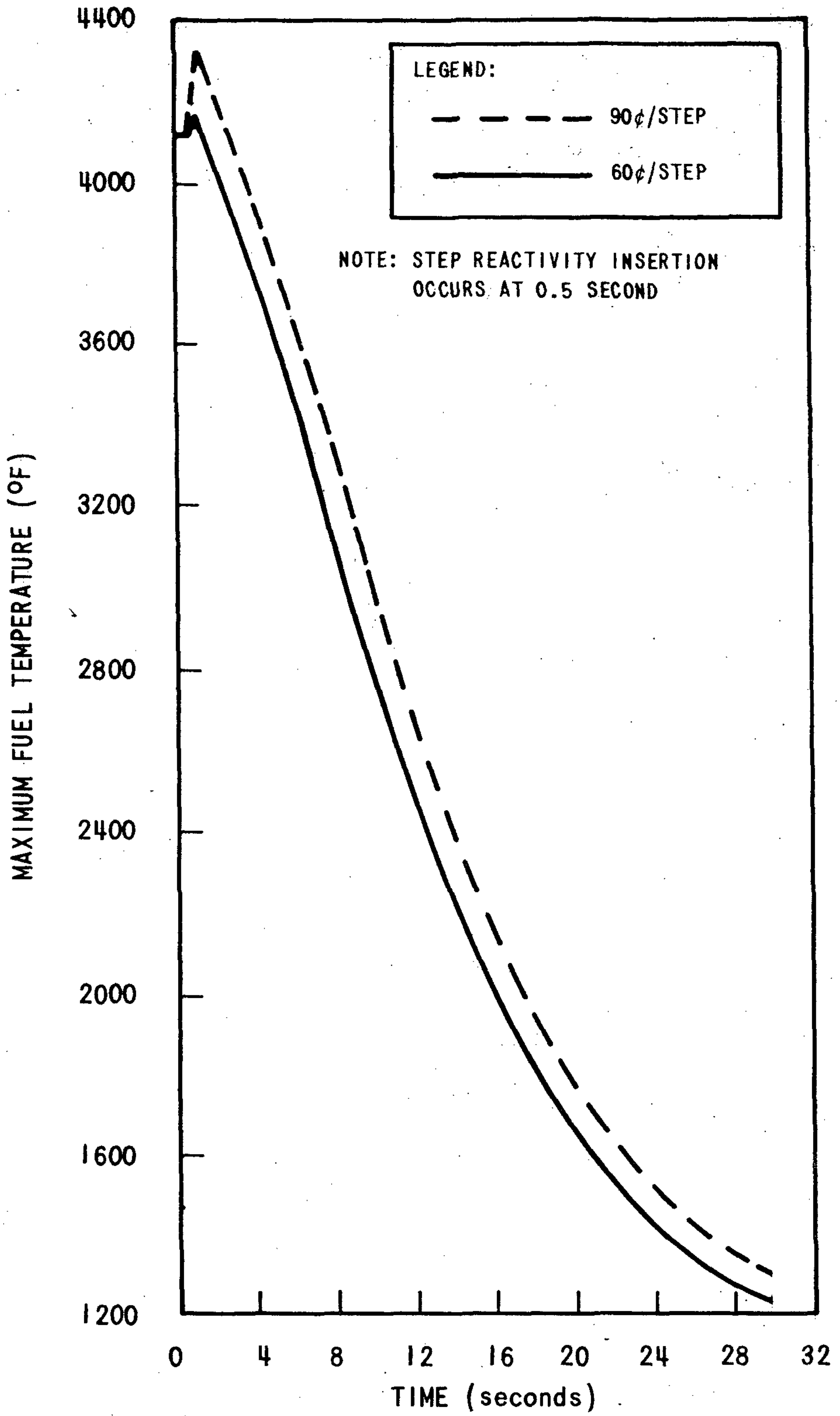

Figure 15.2.3.3-7. Variation in RB/A Hot Channel Maximum Fuel Temperature for SSE Transient

6727-51

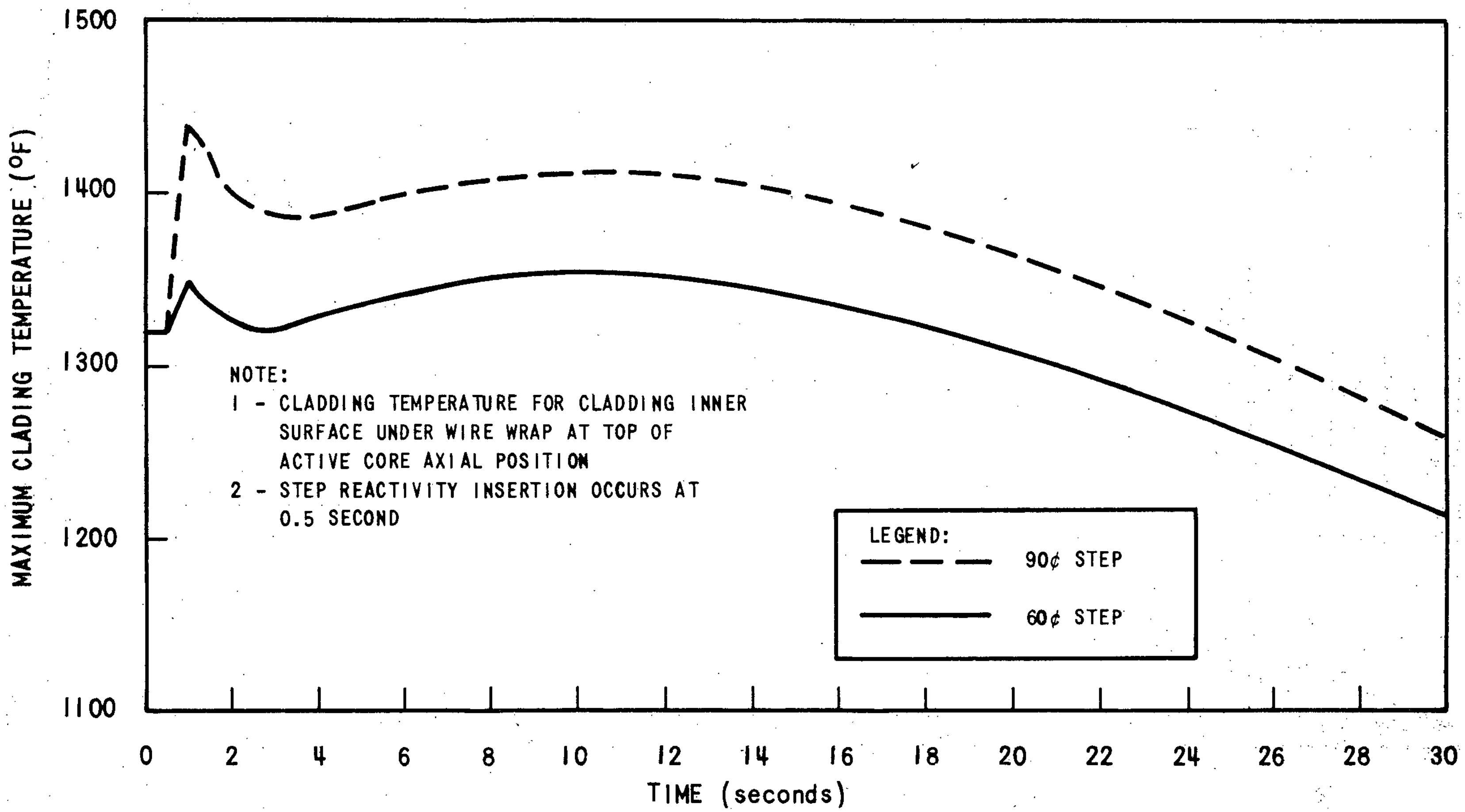

Figure 15.2.3.3-8. Variation in RB/A Hot Channel Maximum Cladding Temperature for SSE Transient

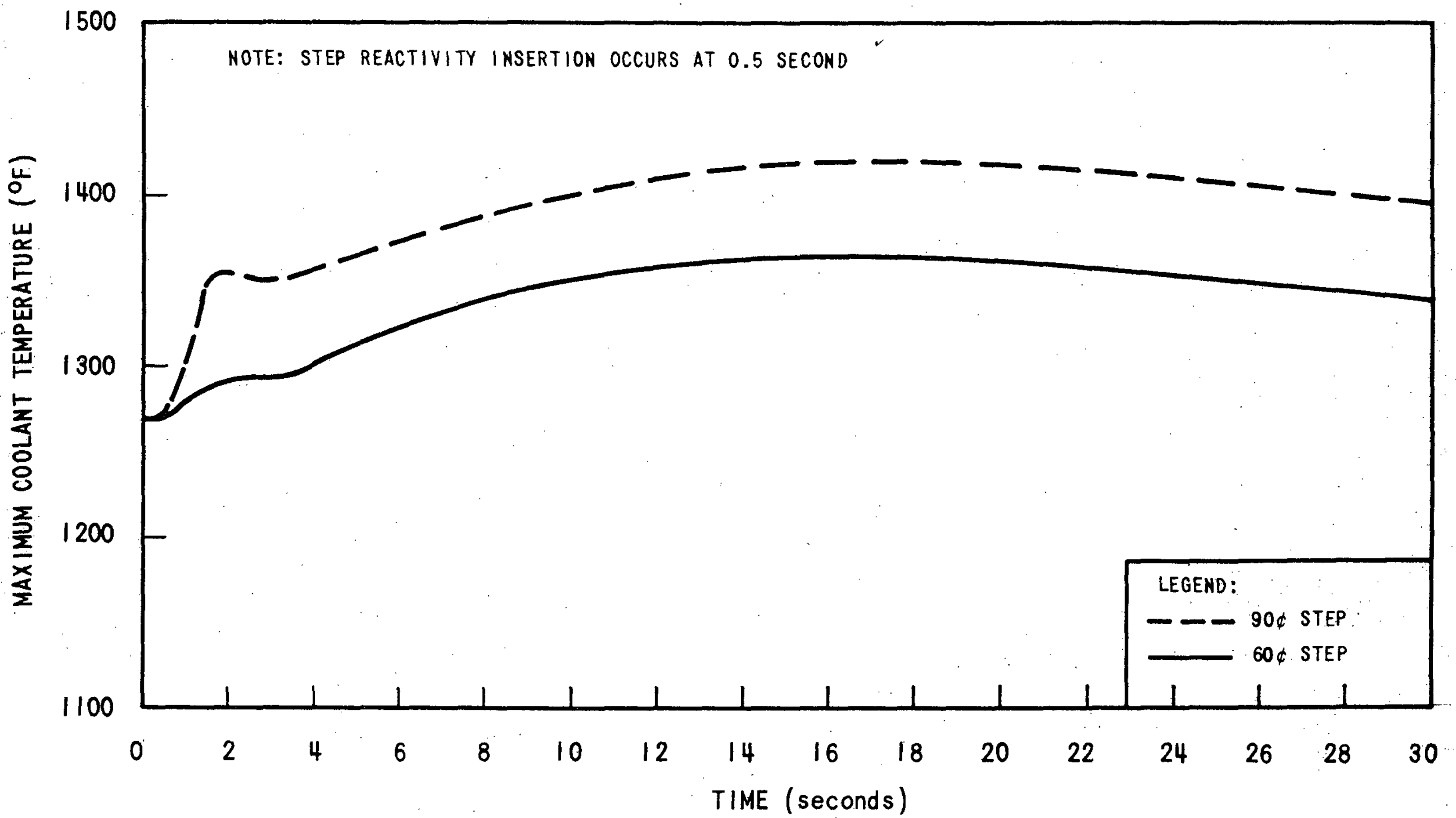

Figure 15.2.3.3-9. Variation in RB/A Hot Channel Maximum Coolant Temperature for SSE Transient

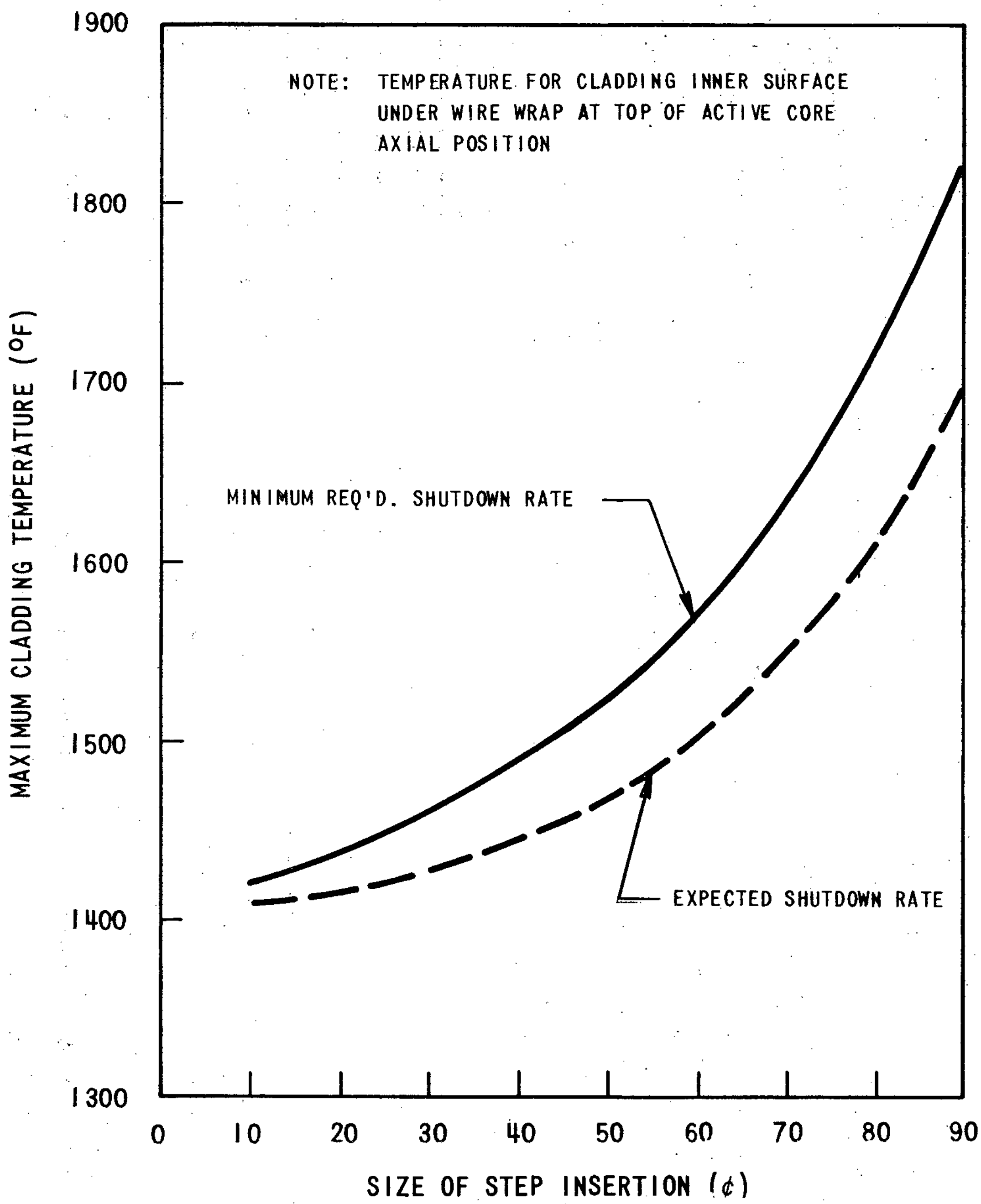

Figure 15.2.3.3-10. Comparison of F/A Hot Spot Cladding Temperature for SSE Using Expected and Minimum Required Primary Control Rod Shutdown Rates

6727-54

15.2.3.4 Control Assembly Withdrawal at Startup - Maximum Mechanical Speed

15.2.3.4.1 Identification of Causes and Accident Description

For this event, it is assumed that the reactor has reached criticality. To reach criticality at any time in the CRBRP core, it is first necessary to completely withdraw the six secondary control rods in Row 7 as well as the two Row 4 primary startup rods (See Section 4.3.2.5). Then to ascend to power requires withdrawal of the remaining primary control rods which are normally sequentially moved to keep them at nearly the same level. The maximum withdrawal speed is about 9 ipm which can cause a maximum reactivity insertion to the core of 2.4¢/sec as discussed in Section 15.2.1.1 (as an anticipated reactor event). A description of the low probability of this event was also given in that section.

Despite the fact that a continuous rod withdrawal incident is very improbable, the event was analyzed assuming additional failures resulting in a withdrawal rate approximately equal to the maximum mechanical speed capability of the CRDM. This maximum rod speed is approximately 73 ipm above which the CRDM roller nut will disengage from the lead screw (due to centrifugal force) resulting in a drop of the rod. To reach a speed near 73 ipm multiple independent failures of the CRDM controller would have to be postulated to produce a very high pulse rate. As discussed in Section 15.2.1.1, the electronic logic circuit that stops the rod when the maximum design withdrawal speed is reached would have to be invalidated. A maximum ramp insertion to the core of 19 /second could occur as the rod passes the core midplane (considering the highest worth control rod). This is an extremely unlikely fault.

15.2.3.4.2 Analysis of Effects and Consequences

To analyze the effects of a continuous rod withdrawal at startup, the reactor was assumed to be initially operating with a very small power generation of 1 Mw, 600°F reactor inlet temperature and 40 percent of full flow. Beginning of equilibrium cycle core conditions were modeled. The minimum Doppler coefficient of -0.005 was used for the core. This value is obtained by decreasing the nominal Doppler coefficient by 20% for uncertainties as discussed in Section 4.3.2.3. Continuous ramp reactivity insertions of 10 and 20 /sec were studied with the FORE-II computer code (see Appendix A).

The secondary control rod system was used for shutdown in the studies (with the maximum worth control assembly assumed to be stuck). Trip was taken at 56 percent full reactor power which adequately accounts for the flux to total flow subsystem performance at this reduced power and flow level. This setting corresponds to a power-to-flow ratio of 1.4. Primary system action would be initiated based on Flux - Pressure at approximately the same time. However, since the response of the primary rods is faster, the resulting transient is less severe. [Note: At full power and low, the seondary control rods would be tripped at a power-to-flow ratio of 1.3].

Figures 15.2.3.5-1 thru -4 show the variation in reactor power and fuel assembly hot channel (with 3σ hot channel factors) maximum temperatures for the fuel, cladding and coolant. A maximum cladding temperature of about 800°F was attained for a 20¢/sec reactivity insertion. As can be seen, higher cladding temperatures are experienced for smaller ramp insertion rates. The 19¢/sec insertion would thus result in slightly higher temperatures. The reason for this effect is that the slower reactivity insertion rates allow more energy to be generated in the core before the trip signal occurs. [Note: Section 15.2.1.1 presents results for even smaller reactivity insertion rates.] Corresponding analysis of the highest power radial blanket assembly hot pin results in temperatures significantly lower than those for the fuel assembly hot pin.

15.2.3.4.3 Conclusions

The extremely unlikely event of a control rod withdrawal at its maximum mechanical speed was analyzed for the core at startup conditions. A maximum cladding temperature of the fuel assembly hot pin of about 800°F was found to result. Since the normal full power maximum temperature at the same position is over 1400°F, the transient should not produce any significant additional degradation of the cladding. A description of how this type event is incorporated in the pin cladding structural design evaluation is given in Section 4.2.1.3.

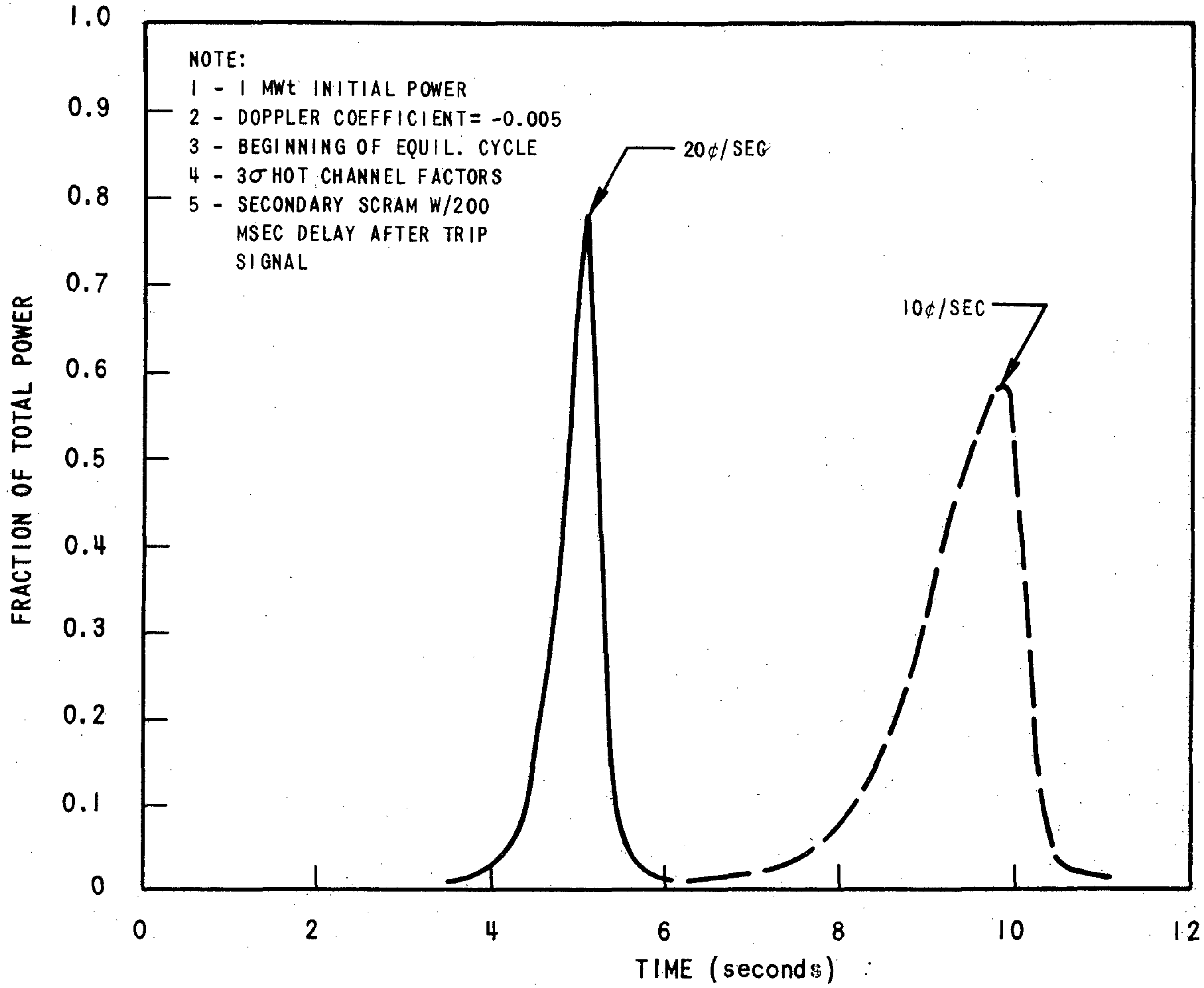

Figure 15.2.3.4-1. Variation of Reactor Power for Control Assembly Withdrawal at Startup

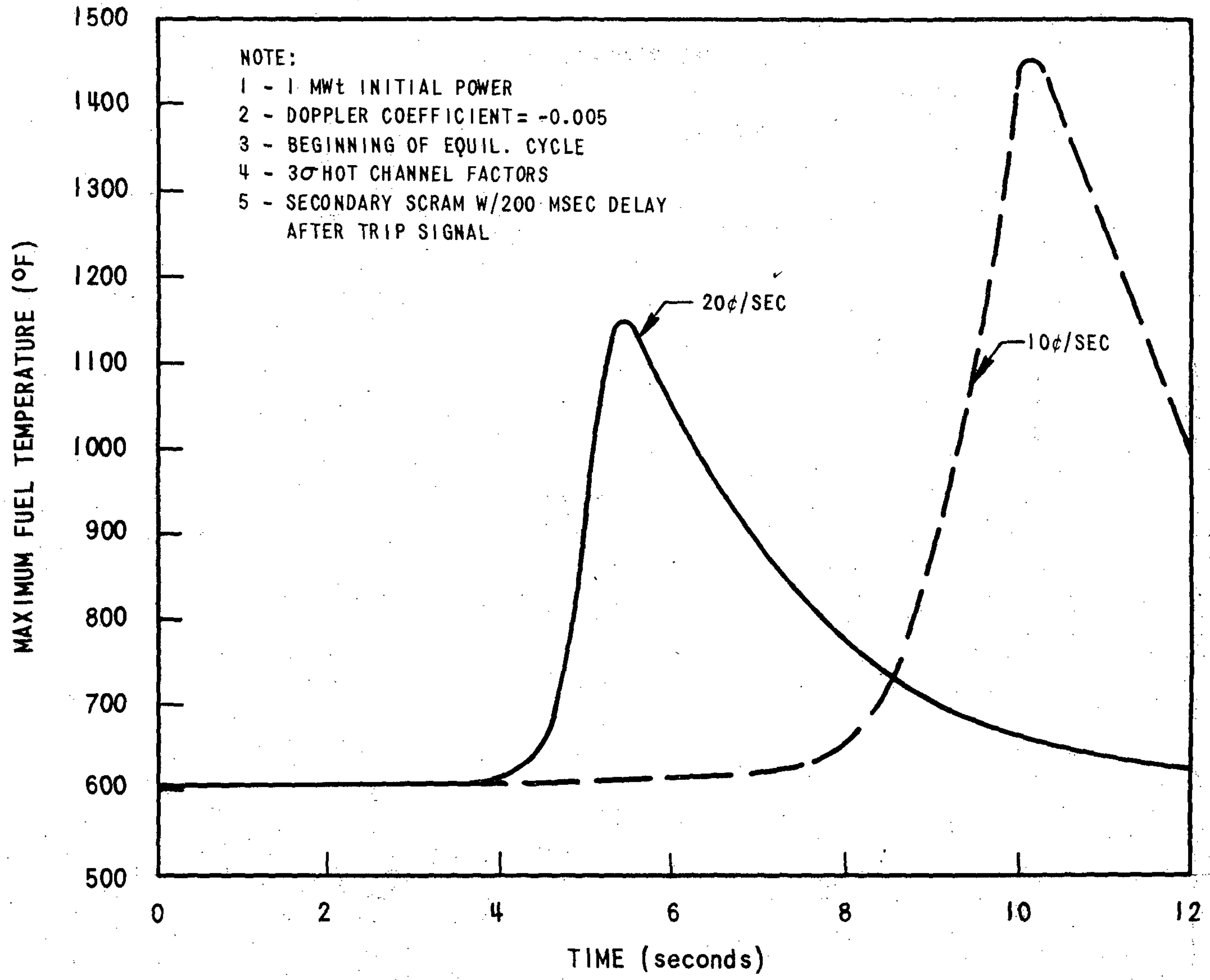

Figure 15.2.3.4-2. Variation in F/A Hot Channel Maximum Fuel Temperature for Control Assembly Withdrawal at Startup

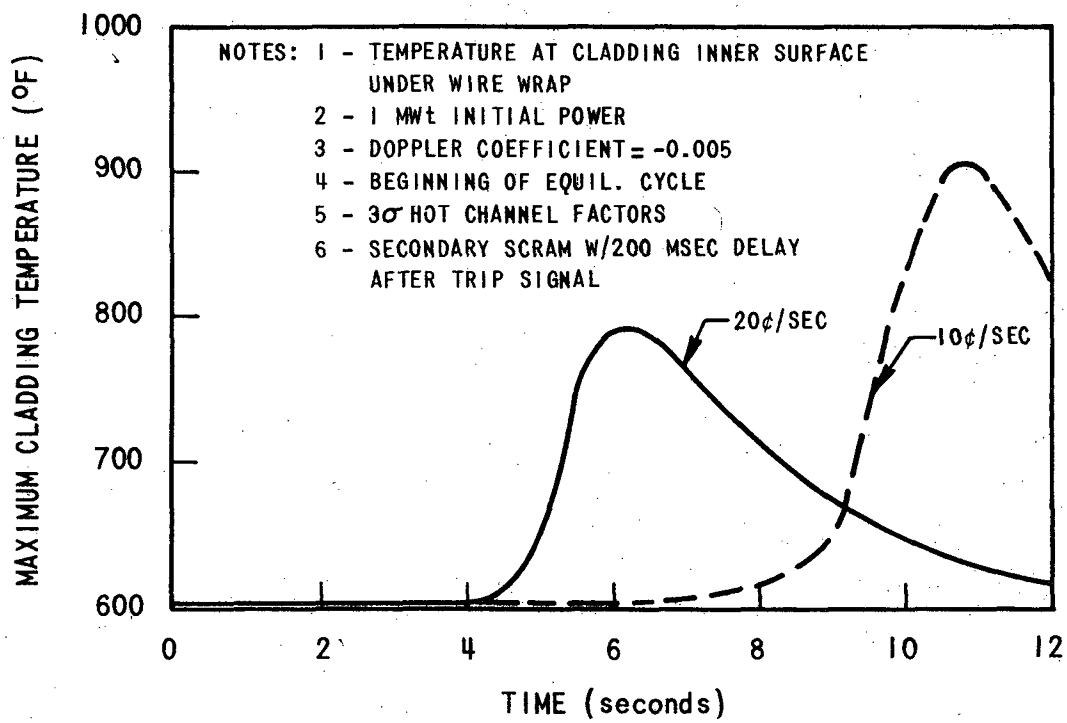

Figure 15.2.3.4-3. Variation in F/A Hot Channel Maximum Cladding Temperature for Control Assembly Withdrawal at Startup

6727-57

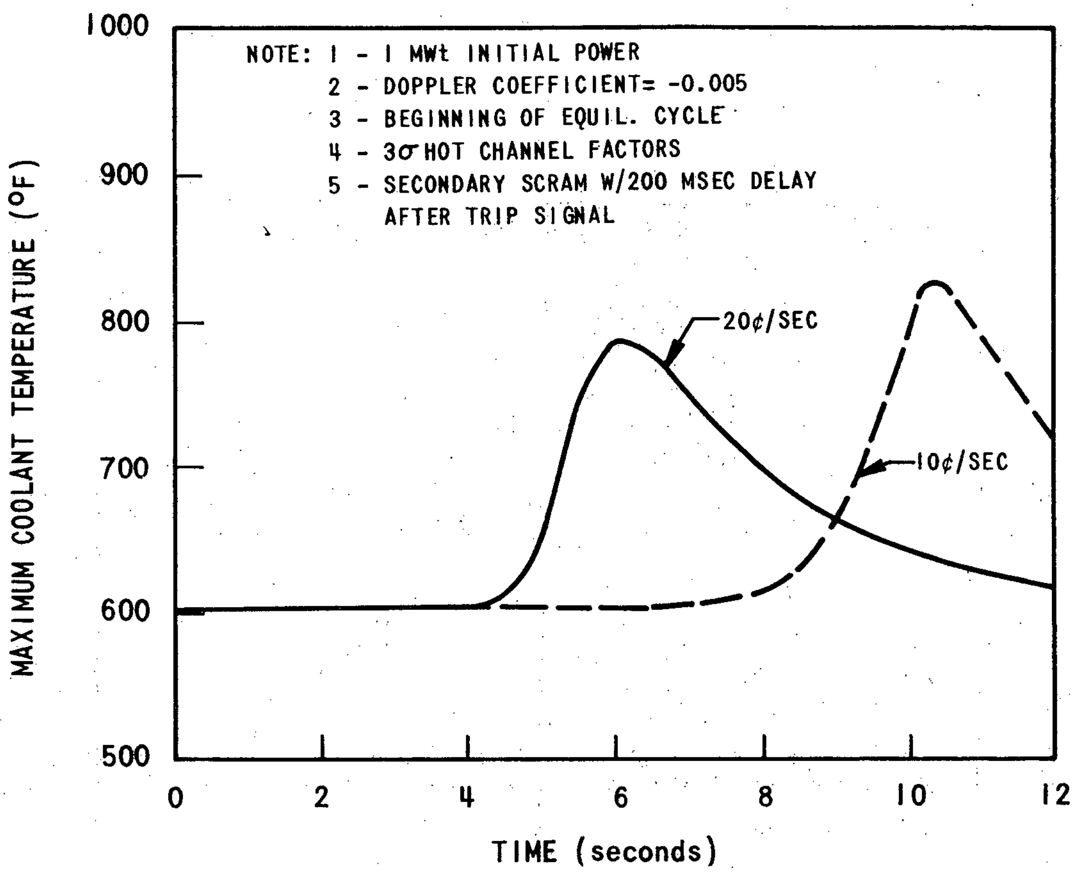

Figure 15.2.3.4-4. Variation in F/A Hot Channel Maximum Coolant Temperature for Control Rod Withdrawal at Startup

6727-58

15.2.3.5 Control Assembly Withdrawal at Power-Maximum Mechanical Speed

15.2.3.5.1 Identification of Causes and Accident Description

During full power reactor operation, the in-core primary control rods can be moved by the automatic control system or manually by the operator. The maximum design withdrawal speed is about 9 ipm which can cause a maximum reactivity insertion to the core of 2.4¢/sec as discussed in Section 15.2.1.2 (as an anticipated reactor event). A description of the inherent safety features employed by the reactor control system to mitigate the consequences of a control assembly withdrawal at power is also presented in Section 15.2.1.2.

Despite the fact that a continuous rod withdrawal incident is very improbable, the event was analyzed assuming additional failures resulting in a withdrawal rate equal to the maximum mechanical speed capability of the CRDM. This maximum rod speed is approximately 73 ipm above which the CRDM roller nut will disengage from the lead screw (due to centrifugal force) resulting in a
51 drop of the rod. To reach a speed near 73 ipm, multiple independent failures of the CRDM controller would have to be postulated to produce a very high pulse rate. Also, as indicated in Section 15.2.1.1, the electronic logic circuit that stops the rod when the maximum design speed is reached would have to be invalidated. A maximum ramp insertion to the core of 19¢/sec could occur as the rod passes the core midplane (the highest worth control rod).

15.2.3.5.2 Analysis of Effects and Consequences

To analyze the effects of a continuous rod withdrawal at power, the reactor was assumed to be initially operating at full power at plant thermal/hydraulic design conditions. Beginning of life and equilibrium core conditions were modeled. The minimum Doppler coefficient (This value is obtained by decreasing the nominal Doppler coefficient by 20% for uncertainties as discussed in Section 4.3.2.3.) of -0.005 was used for the core. Continuous ramp reactivity insertions of 10 and 20¢/sec were studied with the FORE-II computer code (see Appendix A) to simulate a range around the postulated insertions. All operator and automatic corrective actions were neglected and reactor shutdown occurred due to scram.

Both the primary and secondary control rod systems were studied separately for their shutdown capability with the maximum worth control assembly assumed to be stuck. For scram with the primary control system, 15% over-power was used for the trip; scram with the secondary systems used a trip from a power-to-flow ratio of 1.3. A 200 millisecond period was taken for the delay between the trip signal and the beginning of control rod insertion.

Figures 15.2.3.6-1 through -4 show the variation in reactor power and fuel assembly hot channel (with 3σ hot channel factors) maximum temperatures for the fuel, cladding and coolant. As in Section 15.2.1.2, data are given for both primary and secondary control rod system shutdown. The maximum cladding temperature attained for a 20¢/sec reactivity insertion is only about 1420°F for primary scram and 1460°F for secondary scram. As can be seen, slightly higher cladding temperatures can be experienced for smaller ramp insertion rates. The reason for this effect is that the slower rate allows more core energy to be developed before the trip signal occurs. Figure 15.2.3.5-4 indicates, however, that there is very little difference between the maximum cladding temperature for the 10 and 20¢/sec cases. Corresponding analyses of the highest power radial blanket assembly hot pin indicate that its temperature for the 20¢/sec case would be about 100°F cooler than that for the fuel assembly hot pin.

15.2.3.5.3 Conclusions

The extremely unlikely event of a control rod being withdrawn as its maximum mechanical speed was analyzed. A maximum cladding temperature of the fuel assembly hot pin of about 1400°F was found to result for primary scram and 1460°F for secondary scram. Since the normal full power maximum temperature at the same position is over 1400°F, the transient should not produce any significant additional degradation of the cladding. A description of how this type event is incorporated in the pin cladding structural design evaluation is given in Section 4.2.1.3

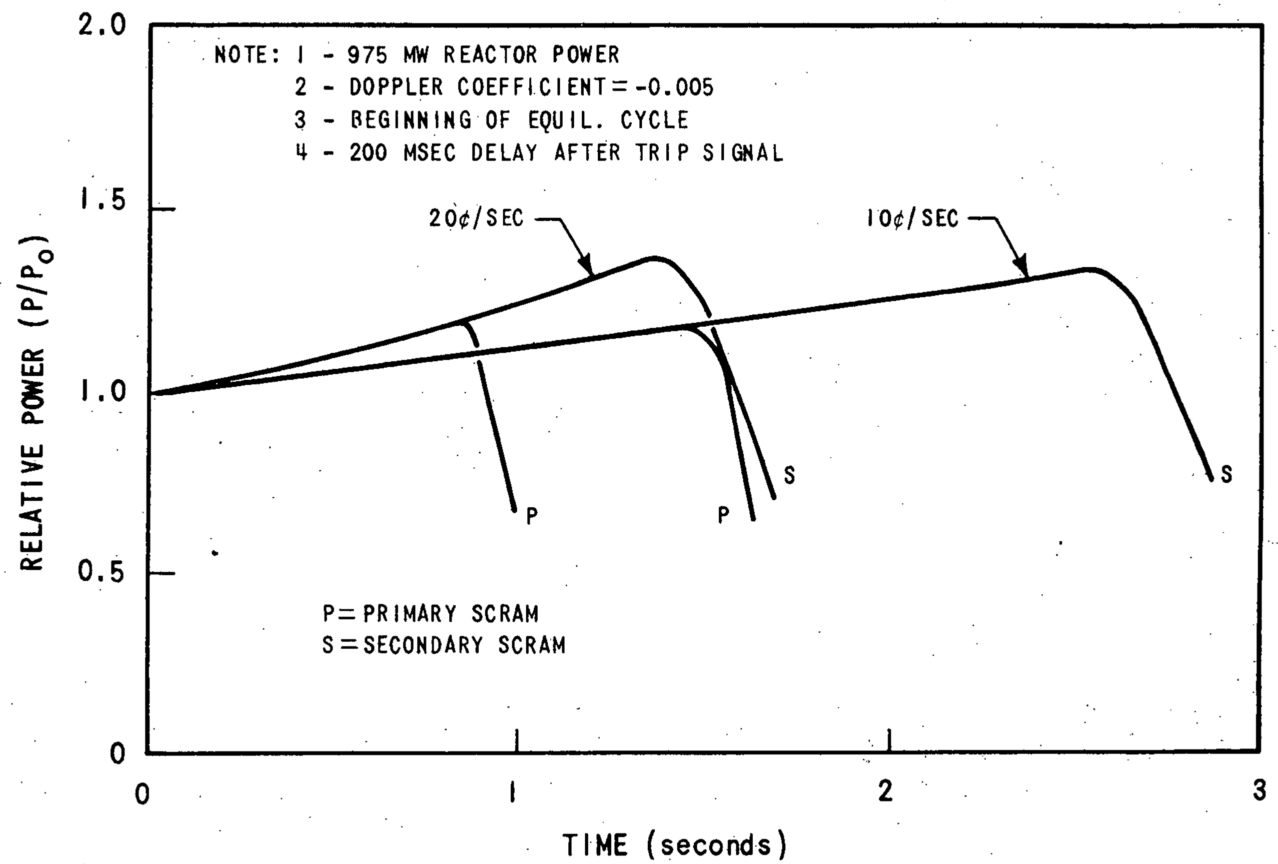

Figure 15.2.3.5-1. Variation in Reactor Power for Control Assembly Withdrawal at Power

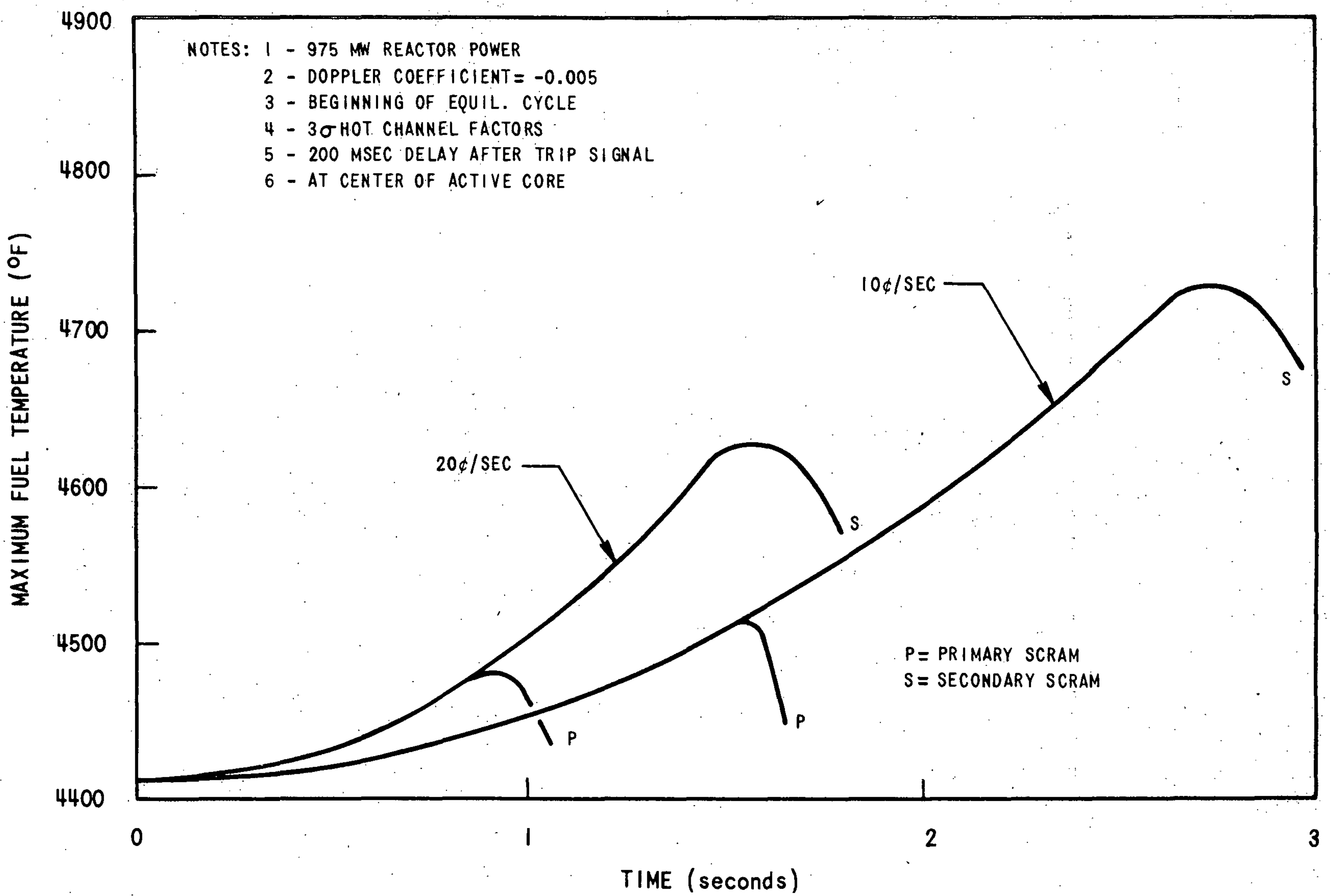

Figure 15.2.3.5-2. Variation in F/A Hot Channel Maximum Fuel Temperature for Control Assembly Withdrawal at Power

6727-60

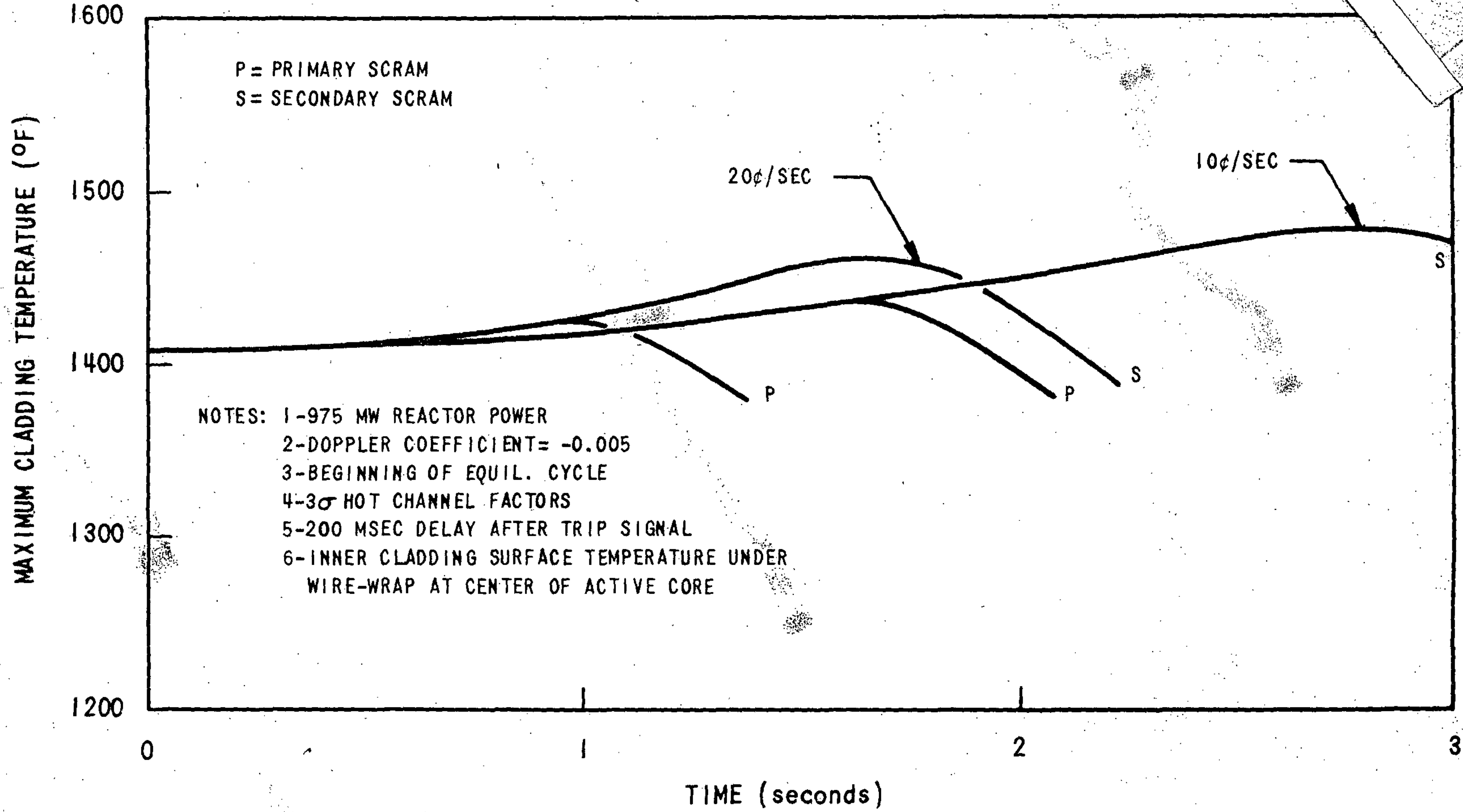

Figure 15.2.3.5-3. Variation in F/A Hot Channel Maximum Cladding Temperature for Control Assembly Withdrawal at Power

6727-62

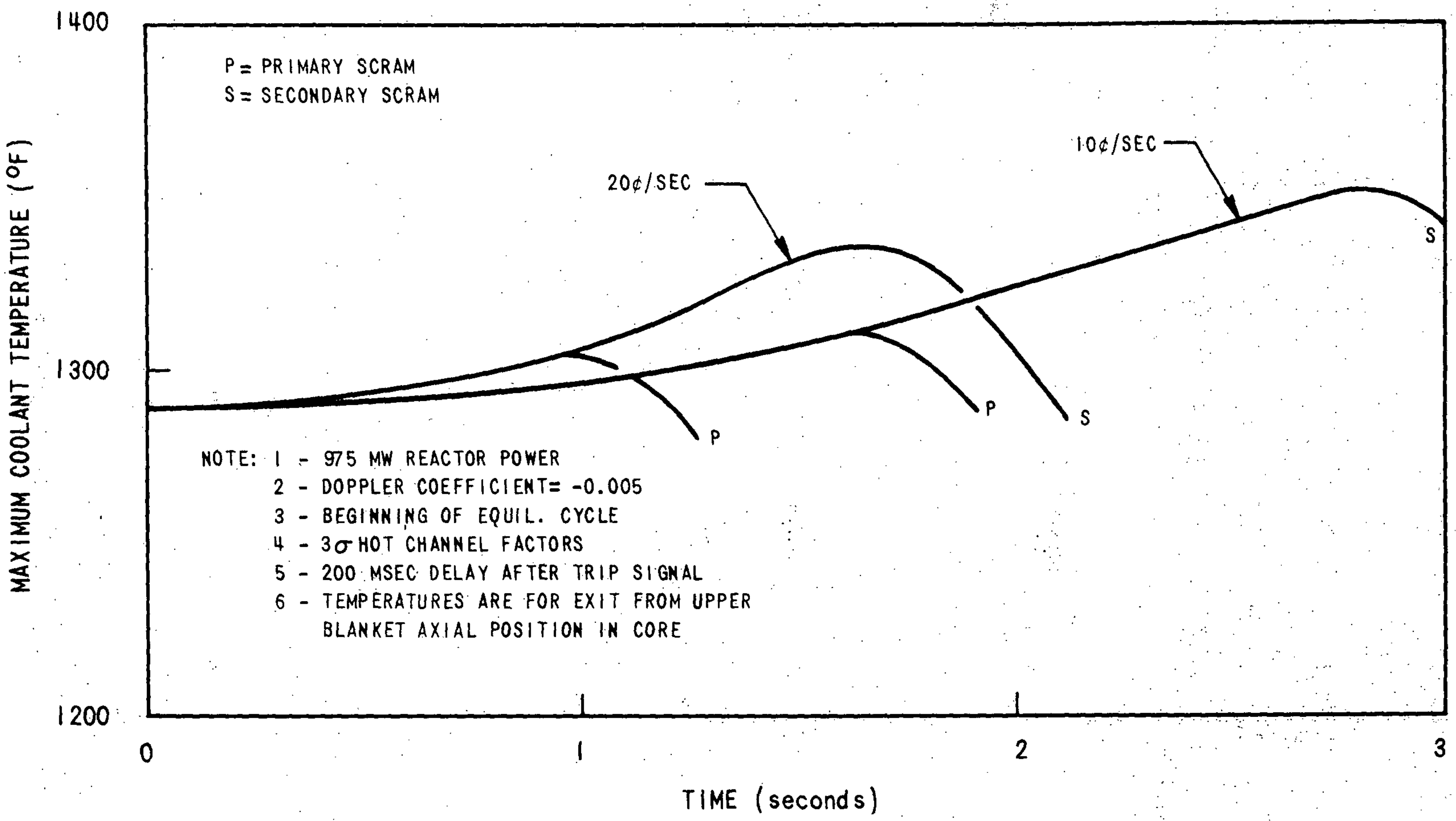

Figure 15.2.3.5-4. Variation in F/A Hot Channel Maximum Coolant Temperature for Control Assembly Withdrawal at P–