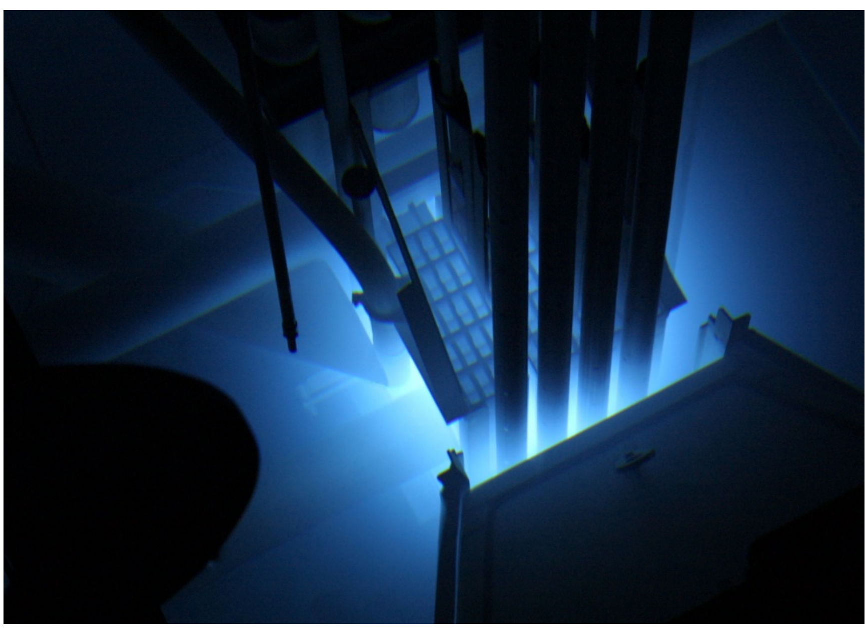
NC STATE UNIVERSITY

SAFETY ANALYSIS FOR ASSESSING 2 MW POWER UPGRADE FOR THE NCSU PULSTAR REACTOR

Nuclear Reactor Program

NORTH CAROLINA STATE UNIVERSITY

RALEIGH, NORTH CAROLINA 27695



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Executive Summary

The objective of this report is to perform design analysis and analyze accident scenarios for the North Carolina State University PULSTAR Reactor in order to evaluate the basis for increasing its nominal steady-state power level from 1 MWt to 2 MWt. In support of this objective, a thermal hydraulic model of the PULSTAR reactor core and cooling system was developed using the RELAP5/MOD3.3 code of the U.S. Nuclear Regulatory Commission, [1] incorporating results of the neutronic analysis previously performed in support of the NCSU License Amendment for the Use of 6% Enriched Fuel. [2]

Utilizing the RELAP model, the design analyses determine the limits of operation for the PULSTAR Reactor beyond which the design criteria are violated. The impacts of variation in core coolant flow, system pressure, and core coolant inlet temperature on the steady-state burnout level, flow stability, and coolant bulk boiling in the hot channel, under both natural convection and forced flow cooling modes of operation were assessed. Additional evaluations were performed for the fuel and cladding temperature criteria under transient conditions, including loss of primary flow, loss of coolant, and reactivity insertion accidents. All analyses were performed following the guidance of NUREG 1537 Parts 1 and 2. [3,4]

The results of the steady-state thermal margins evaluated for forced flow at limiting conditions show that all design criteria are maintained for power levels up to 2 MWt (see Table 4-2), with fuel and cladding temperatures remaining well below accepted safety limits. For the natural circulation mode of operation, results show that the fuel and cladding temperatures remain well below accepted safety limits for power levels up to 1.0 MWt (see Table 4-7).

For transient (i.e. accident) conditions, the results show that fuel and cladding temperatures remain well below accepted safety limits. The following accident scenarios were evaluated:

1. Loss of flow – Scenarios analyzed included a blocked fuel assembly (see Table 4-10), and a complete loss of forced flow with the flapper valve both open and stuck closed (see Table 4-14).
2. Loss of coolant – Scenarios analyzed included three large break locations, leading to partial and full uncovering of the reactor core (see Table 4-19).
3. Reactivity insertions – Scenarios analyzed included ramp and step reactivity insertions initiating at various limiting conditions (see Table 4-24).

The results of these analyses show that the integrity of the fuel will be maintained for all steady-state and transient limiting accident conditions for reactor power levels up to 2 MWt. The safety margin created by adhering to these limits will prevent the reactor design criteria from being exceeded due to abnormal occurrences.

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1. Introduction

The North Carolina State University (NCSU) Pulsed Training Assembled Reactor (PULSTAR) is currently a 1 MWt pool type nuclear research reactor administered by the Nuclear Reactor Program and located in Burlington Engineering Laboratory (BEL) on the NCSU north campus. The NCSU reactor is one of two PULSTAR Reactors built, and the only one still in operation. The other reactor was a 2 MWt reactor at the University of Buffalo, which went critical in 1964 and was decommissioned in 1994. The NCSU PULSTAR is shown in Figure 1-1 (also, see Figure 1-5 for an elevation view).

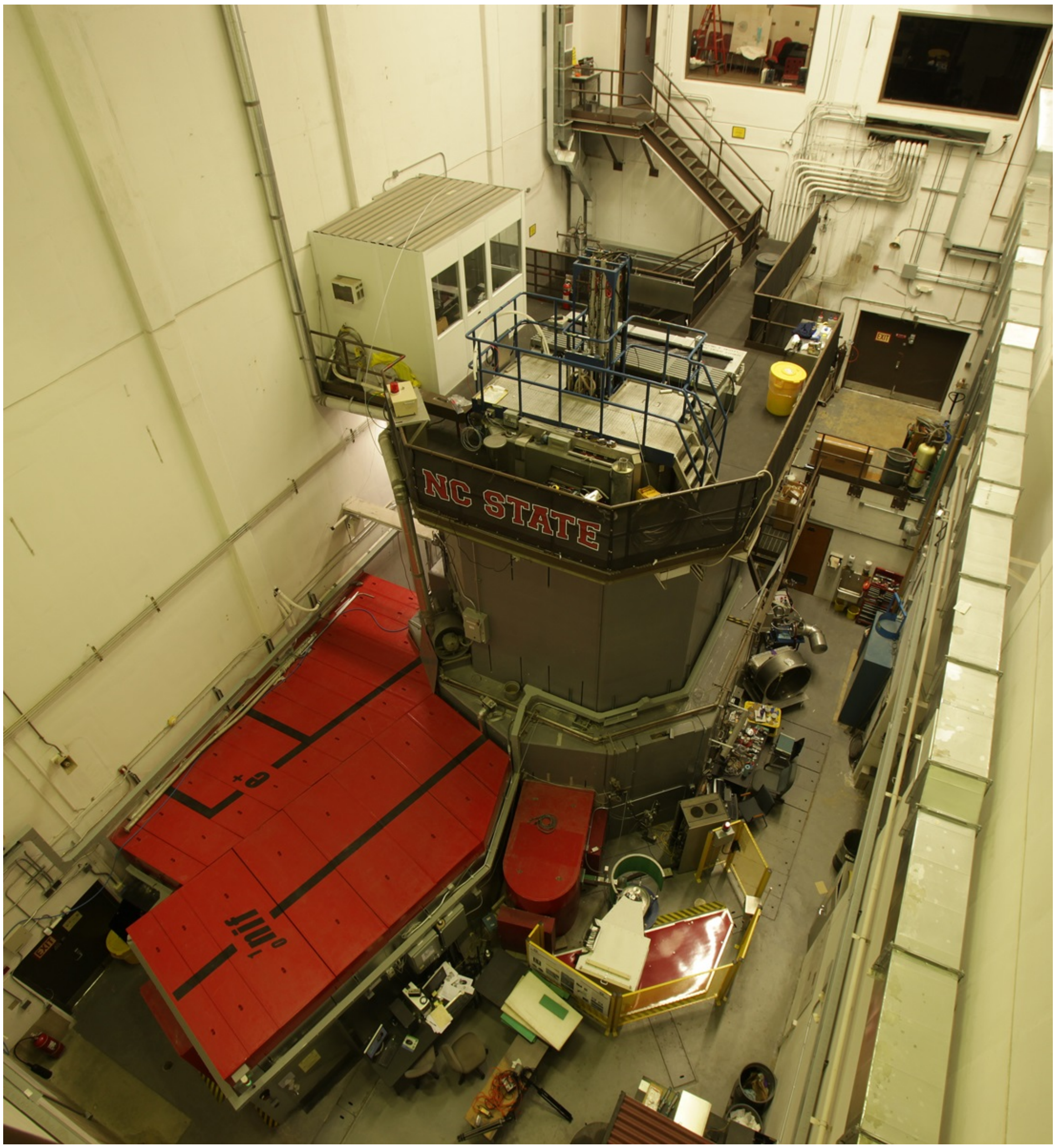

Figure 1-1 – NCSU PULSTAR Reactor Facility

1.1. Objective

This report describes the analyses of steady-state conditions and accident scenarios to evaluate increasing the PULSTAR steady-state licensed power from 1 MWt to 2 MWt. The following accidents were analyzed:

- Loss of Primary Flow
- Loss of Coolant
- Reactivity Insertion

The analyses were performed using the RELAP5/MOD3.3 code of the U.S. Nuclear Regulatory Commission (NRC). [1] This report describes the RELAP5 model of the PULSTAR systems, steady-state initializations, and results of the accident analyses.

The analysis in this report has been generated following the guidance and requirements of NUREG 1537 Parts 1 and 2, [3,4] applicable sections of 10 CFR 50 *Domestic Licensing of Production and Utilization Facilities*, NUREG/CR-5535[1], and utilizes the results of neutronic analysis performed in support of the NCSU *License Amendment for the Use of 6% Enriched Fuel.* [2]

1.2. Overview of PULSTAR Design Features

The North Carolina State University PULSTAR Reactor was manufactured by the American Machine and Foundry Company (AMF). The design, fabrication, and installation were based on the proven prototype located at the Buffalo Materials Research Center (BMRC) at the State University of New York at Buffalo. The Buffalo PULSTAR operated at a power of 2 MWt from 1964 to 1995 using uranium dioxide (UO_2) fuel that was 6% enriched in ^{235}U and reached burnup limits exceeding 15,000 MWD/MTU. The NCSU PULSTAR is a light water moderated and cooled thermal reactor that uses UO_2 fuel enriched to 4% or 6% in ^{235}U. Initial criticality of the PULSTAR occurred in September 1972. It is operated by the Nuclear Reactor Program (NRP) in the Department of Nuclear Engineering within the College of Engineering. The NCSU PULSTAR currently operates at steady-state power levels of up to 1 MWt. The NCSU PULSTAR was originally designed to be pulsed routinely to 2200 MW peak power with a 38 MW-sec total energy release. However, it no longer operates in pulse mode.

The NCSU PULSTAR core is a heterogeneous system of light water and UO_2 fuel, which provides a source of neutrons for research purposes. The core is immersed under a nominal twenty feet of water in an open pool. The reactor pool is lined with aluminum and surrounded on the sides and bottom by concrete shielding.

Control of the power level is accomplished by the variable positioning of neutron absorbing rods within the core. These control rods as well as the various neutron detecting chambers used for power level measurement, are suspended from a bridge which spans the reactor pool. The reactor is operated from a console located in a control room visible from the bridge.

The NCSU PULSTAR has a number of design features which make it an extremely safe research reactor. The first of these features is the use of low enriched UO_2 in sintered pellet form as fuel. This fuel provides an inherent safety shutdown feature due to the strong negative temperature feedback (Doppler) effect. The heat capacity of UO_2 is quite large, permitting a large release of energy in the core under transient conditions without exceeding the melting point of the cladding. The low thermal diffusivity of UO_2 leads to a long thermal time constant for the fuel (approximately 4 seconds). This long time constant prevents

the explosive formation of steam experienced in plate-type metallic reactors undergoing severe reactor transients. The time constant for a typical plate-type aluminum alloyed fuel assembly for instance, is on the order of a few milliseconds.

Considerable experience and information has been gathered on the characteristics of UO_2 under irradiation conditions. The chemical and radiation stability of UO_2 is known to be excellent. The ability of UO_2 to retain fission products is also excellent and provides a strong motivation for the use of UO_2 aside from the obvious advantages in performance. Use of the fuel in pellet form ensures against the rapid release of energy which could occur through loss of clad integrity and the subsequent dispersion of fuel into the coolant if the fuel were in powder form.

The nuclear power industry has generated a significant quantity of information on the properties of zircaloy and on operating experience with zircaloy clad UO_2 fuel. Therefore, the clad material is considered a proven material for reactor use. The high melting temperature of zircaloy may be considered a distinct advantage for use in a reactor core.

One of the strongest arguments for the inherent safety of the PULSTAR core is its similarity to the SPERT oxide core, which has undergone extensive testing. [5] Transients releasing as much as 100 MW-sec of energy have been initiated in the SPERT core without causing damage. The only serious defect discovered in SPERT tests was the presence of a double peaked pulse caused by coherent bowing of the fuel pins during a transient. The coherent bowing was eliminated when the pins were supported in a fashion that provided an 18-inch unsupported length of pin instead of the unsupported length of 6 feet. The PULSTAR fuel is designed to provide an unsupported length of only eight inches so that coherent bowing is not a problem. These supports also serve as wear surfaces to prevent damage to the fuel pin cladding.

1.3. Description of Reactor Core and Coolant Systems

1.3.1. Reactor Core and Pool

The reactor, shown in Figure 1-2, consists of the fueled core, reflector assemblies, control rods and guides, the grid plate support structure, and the lower cooling plenum chamber. The core is comprised of up to 25 fuel assemblies of either 4% or 6% enrichment, reflected by up to 10 beryllium or graphite reflector assemblies, and operated by four control rods comprised of silver, indium and cadmium. An analysis of the neutronic parameters associated with the operation of the mixed enrichment core is given in the *License Amendment for the Use of 6% enriched Fuel-Appendix A: Examination of Mixed Enrichment Core Loading for the NCSU PULSTAR Reactor.*[2] As shown in Figure 1-4, beam tubes and a thermal column provide access to neutrons for experiments and other activities that require neutron activation for analysis.

The reactor is immersed under a nominal twenty feet of water in an open pool. The reactor pool stores a large quantity of demineralized water under atmospheric conditions and provides thermal inertial and heat removal for both normal operation and accident conditions. During normal operation, pool water flows downward through the core by forced convection and core heat is transferred through the heat exchanger and rejected to the atmosphere through the cooling tower. Under accident conditions, the flapper valve located at the side of the lower plenum opens by loss of differential pressure to provide long-term stable cooling of the core by natural convection.

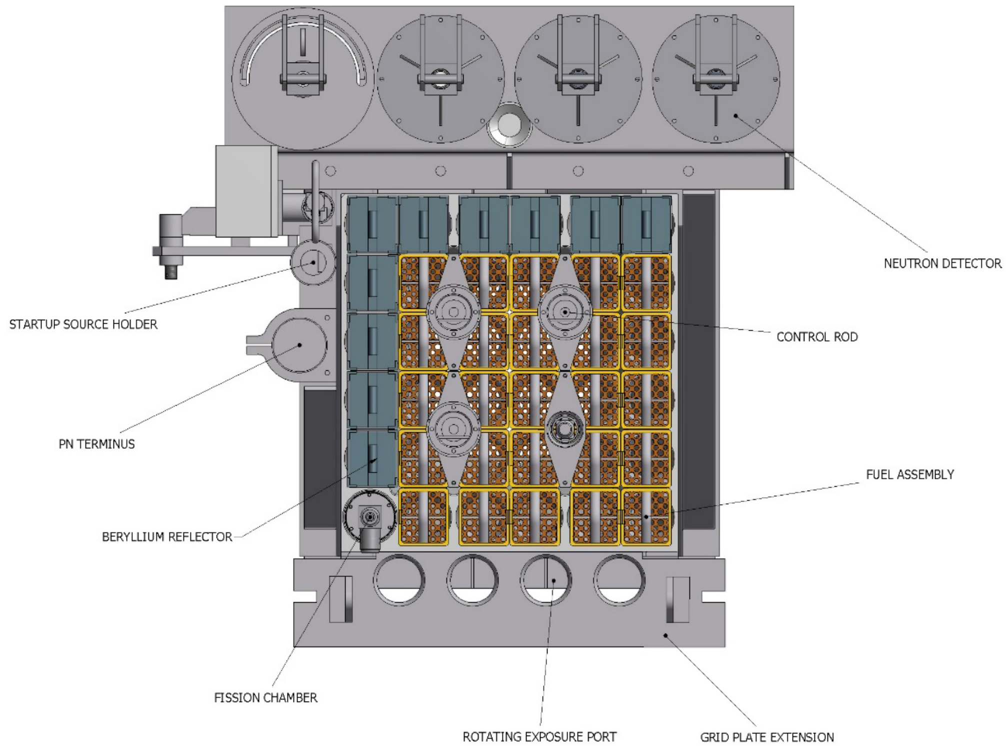

Figure 1-2 – PULSTAR Core Layout

1.3.2. Coolant Systems

The reactor coolant system is designed to remove in excess of two megawatts of heat from the PULSTAR Reactor operating in the steady-state mode with forced convection cooling and has sufficient capacity to operate at power levels up to 100 kW under natural convection cooling. The reactor coolant system also has the ability to cool the reactor during a flow reversal transient should the primary coolant pump fail.

The primary coolant system is an open pool loop containing approximately 15,000 gallons of demineralized water at atmospheric pressure and is designed to maintain a core inlet temperature of 105°F with a 1000 gpm flow rate. The heat from fission is absorbed by the coolant as it flows downwards through the reactor core and is transferred to the secondary coolant system through a plate and frame type heat exchanger. The heat is then removed from the secondary coolant system by a standard counter-flow evaporative cooling tower. The primary and secondary coolant systems and their supporting components are shown schematically in Figure 1-3.

During forced convection operation, the primary water flows downward from the reactor pool through the reactor core and into the core plenum. The plenum is bolted to the 10-inch primary coolant outlet pipe at the bottom center of the pool liner and during forced convection, directs the flow of water from the reactor core to the coolant outlet pipe. The plenum serves as a transition from the square reactor core grid to the round reactor outlet pipe, and as the mechanical support for the reactor core. The plenum is 3 feet high, 22 inches by 20.5 inches at the top, and 10 inches in diameter at the bottom. A flapper valve on the side of the plenum, which is a 15.75-inch diameter flat disc, is manually shut prior to initiation of forced flow and is held shut by the differential pressure created by the downward flow of the coolant

through the plenum. The flapper is counter-balanced to fall open by 30° due to gravity upon loss of forced flow, allowing the transition from downward forced convection to upward natural circulation cooling. The flapper valve is manually operated and closed by means of a reach rod from the reactor bridge area.

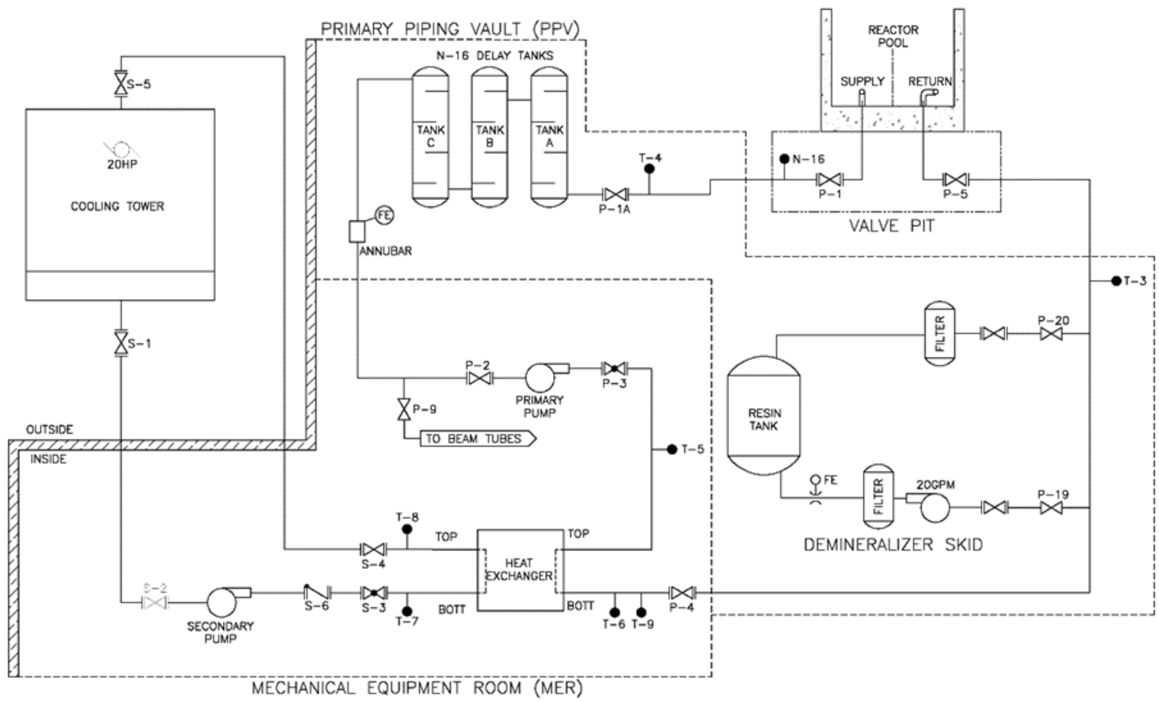

Figure 1-3 – PULSTAR Coolant System Layout

The primary coolant exits the plenum by means of a 10-inch stainless steel pipe, passing through the pool liner and through a 10-inch manual isolation valve (P-1) in the valve pit. All of the stainless steel pipe that is embedded in concrete is bituminous coated and felt wrapped to prevent corrosion. The coolant passes through a tunnel, through a second 10-inch manual isolation valve (P-1A), and then onto the ^{16}N delay tanks located in the Primary Piping Vault (PPV). The PPV is a heavily shielded below grade structure at the south side of the reactor building. The ^{16}N tanks, with a total nominal volume of 1350 gallons, are constructed of stainless steel and have internal baffling to delay the transit of the primary coolant for approximately 1½ minutes at a 1000 gpm flow rate. This 1½ minute delay allows for the ^{16}N activity produced as the coolant flows through the core to decay to near background levels before it exits the PPV. The high energy gamma rays associated with the decay of ^{16}N are shielded by the concrete and earth surrounding the PPV. To allow for system venting and draining, the delay tanks have manual drain and vent valves.

After the coolant leaves the PPV it enters the Mechanical Equipment Room (MER) where it passes through another 10-inch manual gate isolation valve (P-2). The 10-inch pipe is reduced to an 8-inch pipe and enters the suction side of the primary pump. The primary pump is a single stage horizontal centrifugal pump constructed of stainless steel and provides flow at 1000 gpm with a discharge pressure of 12.0 psig. The

speed of the pump is manually set with a variable frequency drive (VFD) to provide a constant flow rate of 1000 gpm.

Exiting the primary pump, the coolant passes through an 8-inch manual throttling globe valve (P-3) and into the counter-flow plate type heat exchanger. The heat exchanger has 126 plates with the capability for additional plates for increased cooling capacity. All wetted parts in the heat exchanger are constructed of stainless steel.

Exiting the heat exchanger, the primary piping diameter expands back to 10 inches and coolant passes through another isolation gate valve (P-4). The coolant exits the Mechanical Equipment Room and re-enters the PPV. A small portion of the coolant is shunted to the primary demineralizer for purification. The primary coolant (cold leg) then enters the tunnel parallel to the hot leg and travels through the tunnel to a 10-inch manual gate isolation valve (P-5) before re-entering the reactor pool. The coolant is discharged into the bottom of the reactor pool through a 90° elbow which directs the water away from the core where it mixes with the bulk of the coolant.

The temperature of the pool is a nominal 105°F when operating at two megawatts, with the temperature rise across the core being 13.8°F. This results in a nominal reactor core coolant outlet temperature of 118.8°F. The reactor pool temperature is controlled by regulating the temperature of the secondary coolant via a variable frequency drive (VFD) on the cooling tower fan motor.

In the natural convection mode of operation, the primary pump is secured and forced flow ceases. The cessation of flow through the reactor outlet plenum results in a loss of the differential pressure across the flapper valve, and the flapper then falls open due to the force of gravity. Water from the pool can now enter the outlet plenum through the open flapper valve and flow upward through the reactor core by thermal convection, thus cooling the reactor.

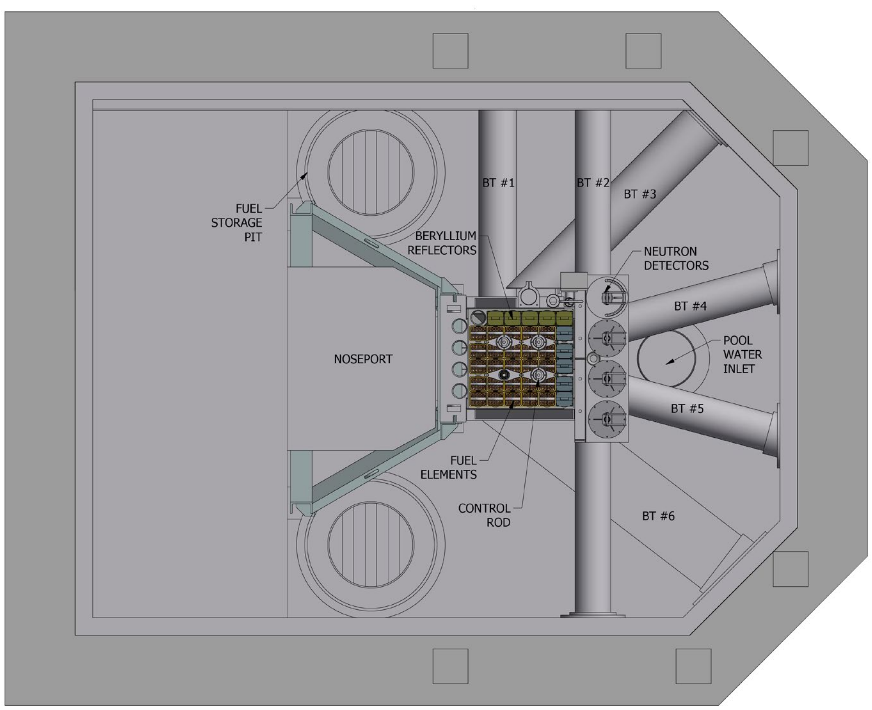

Figure 1-4 – PULSTAR Pool Plan View

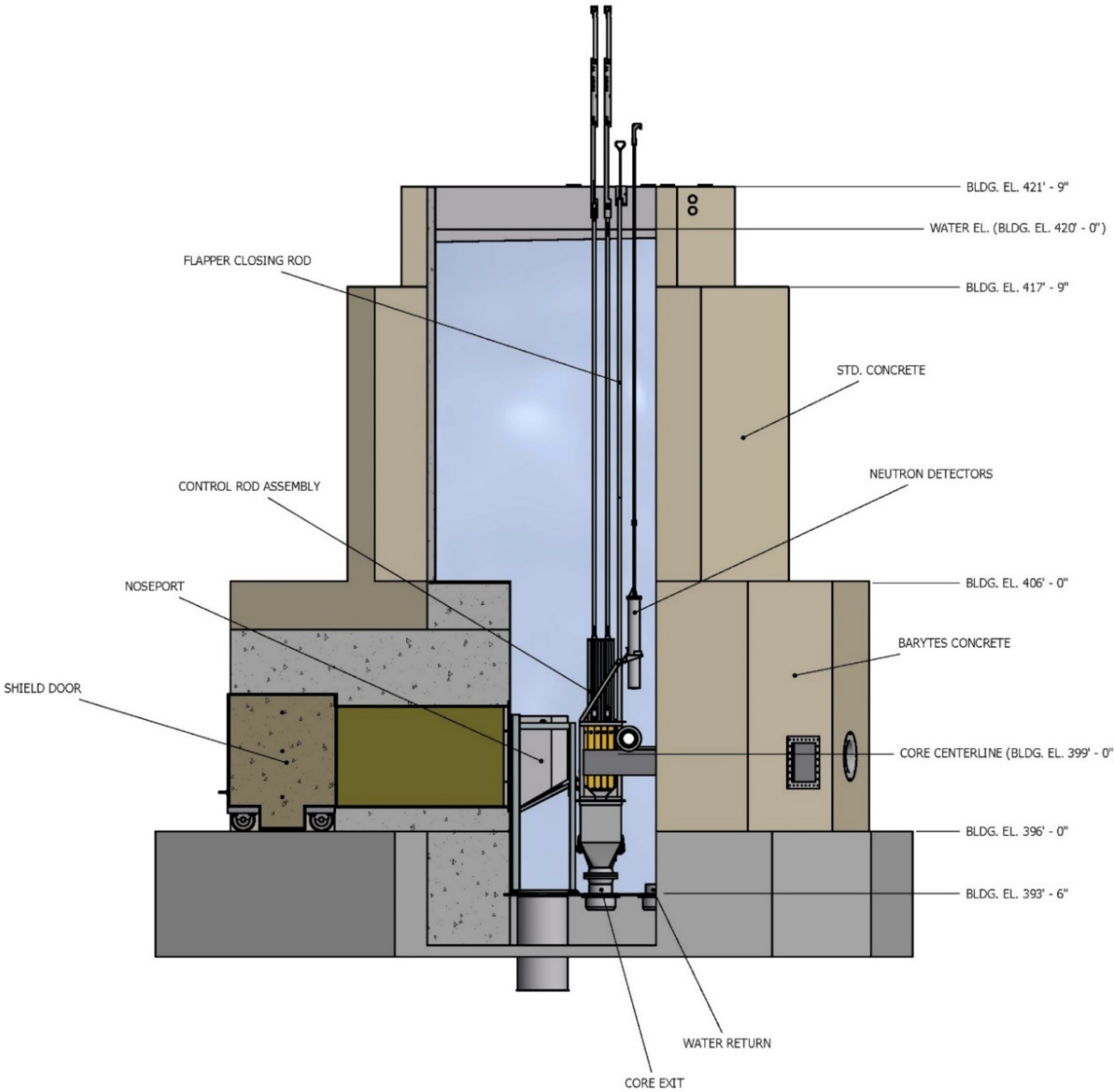

Figure 1-5 – NCSU PULSTAR Elevation View

2. PULSTAR Thermal-Hydraulic Characteristics

2.1. Steady-state Analysis

Thermal hydraulic analyses have been performed to verify that the PULSTAR design criteria will not be exceeded under steady-state or accident transient conditions. The objective of these analyses is to provide bases for determining the license limits.

2.1.1. Design Criteria

The steady-state heat transfer design of the NCSU PULSTAR reactor is based on meeting five criteria:

No Bulk Boiling Criterion	Under forced convection cooling with downward flow, no coolant bulk boiling is allowed in any channel.[3,4]
Flow Instability Criterion	There should be no coolant flow instability in any fuel channel that could lead to a significant decrease in fuel cooling.[3,4]
DNBR Criterion	The ratio of the calculated heat flux at the point of departure from nucleate boiling (DNB) to the maximum steady-state heat flux is greater than 2.0.[4]
Fuel Temperature Criterion	The maximum temperature of the fuel is less than 4352 °F.[3]
Cladding Temperature Criterion	The maximum temperature of the cladding is less than 2200 °F.[a]

The criterion for selecting a safety limit is to ensure the integrity of the fuel cladding. The interrelated variables associated with the core thermal and hydraulic performance with forced convection flow are:

P Reactor thermal power

W Reactor primary coolant flow rate

H Height of water above the top of the core

T Reactor primary coolant inlet temperature

[a]Maximum cladding temperature in NUREG 1537 Part1 Appendix 14.1 is listed as 1500 °C (2732 °F) for PULSTAR fuel. The cladding temperature limit of 1200 °C (2200 °F) along with a 17% cladding oxidation limit was set based on discussion with the NRC via teleconference followed up by an email on August 1, 2018.

When all values are jointly maintained with the limits determined by the safety analysis, fuel cladding integrity will not be lost. The safety limits preclude flow instabilities in the hottest channel and ensure that minimum steady-state DNB ratio is at least 2.0.

2.1.2. Design Objectives

The thermal hydraulic design calculations performed for the NCSU PULSTAR Reactor have an objective which is to determine the limits of operation beyond which the design criteria given above would be violated. The safety margin created by adhering to these limits will protect the reactor from abnormal occurrences causing the design criteria to be exceeded.

The important operating parameters with respect to the thermal hydraulics of the core are the power level, primary coolant flow rate, coolant channel inlet temperature (i.e., bulk pool water temperature) and coolant pressure as determined by the pool water height above the top of the core. A complete thermal hydraulic analysis of the core therefore requires that these operating parameters be examined to determine their worst or most adverse operating limit since this will influence the design limits of the core.

For the NCSU PULSTAR thermal hydraulic analysis, the key independent process variable was chosen to be power level. In other words, the parameters of flow, pressure, and coolant channel inlet temperature were established and then the power level was varied until the design limit was reached. The most adverse values of coolant temperature, pressure and flow rate were chosen as initiating conditions in the analyses. The objective is to present an analysis which provides a conservative design basis for the PULSTAR under all steady-state and transient conditions of operation.

The core thermal hydraulic design provides bases required for the Technical Specifications. Therefore, these analyses are the framework for determining the core Safety Limits (SL) and Limiting Safety System Settings (LSSS) as presented in the Technical Specifications.

2.1.3. Design Evaluation

The NCSU PULSTAR can operate in two steady state primary coolant flow modes, forced convection and natural circulation. A description of the reactor coolant system, and its operation under both cooling modes, is given in Section 1.3.2 above. The steady-state conditions analyzed in this report are summarized below.

2.1.3.1. Natural Convection Mode

A detailed thermal hydraulic analysis has been performed to determine the operating limits under natural circulation conditions. Under natural circulation conditions, bulk boiling can occur resulting in undesirable releases of ^{16}N activity following possible bubble rise in the pool water, although bubble collapse would occur at some level due to sub-cooling. To eliminate this possibility, a limiting criterion for operation is established such that no bulk boiling is allowed. The full thermal hydraulic analysis of natural circulation cooling for the PULSTAR is presented in Section 4.2.2.

2.1.3.2. Forced Convection Mode

A detailed thermal hydraulic analysis was performed to determine the operating limits under forced convection conditions. The objective of the analysis was to determine the effects of variation in core

coolant flow, system pressure, and core coolant inlet temperature on the steady-state burnout level, flow stability, and coolant bulk boiling in the hot channel of the core. The full thermal hydraulic analysis of forced convection cooling for the PULSTAR is presented in Section 4.2.1.

2.2. Transient Analysis

2.2.1. Design Criteria

The transient heat transfer design of the NCSU PULSTAR reactor is based on meeting two criteria:

Fuel Temperature Criterion	The maximum temperature of the fuel is less than 4352 °F.[3]
Cladding Temperature Criterion	The maximum temperature of the cladding is less than 2200 °F.[a]

In addition, cladding oxidation was analyzed for loss of coolant accidents and was found to be below 17% in all cases. The accident transients analyzed in this report are summarized below. The full thermal hydraulic analyses of PULSTAR transients is presented in Section 4.3.

2.2.2. Loss of Flow Accident

A loss of flow accident (LOFA) resulting from a primary pump trip at the NCSU PULSTAR reactor would lead to core flow reversal since the downward flow would eventually be terminated under the influence of a natural circulation driving head. The reactor scrams automatically upon receiving a loss of flow signal. It is conservatively assumed that the primary pump stops instantaneously at the time of loss of flow.

At the initiation of the transient, power-cooling mismatch heats up the core coolant, which in turn decreases the core power by negative reactivity feedback before the reactor scram. The heat generation in the fuel is significantly reduced following the reactor scram. Transition from nominal downward flow to upward natural circulation flow occurs when the core power decreases to a decay heat level. Heat input from the fuel following the scram is low enough that the core safety is ensured by natural circulation cooling. The analysis shows that maximum cladding temperatures remain well below the limit, even in the scenario where the flapper valve fails to open and remains closed. The LOFA transient is discussed in detail in Section 4.3.1.2 of this report.

2.2.3. Blocked Flow Accident

A blocked flow accident could occur if the local flow to a fuel assembly at the core coolant inlet is obstructed due to the presence of a foreign object on top of the core. The PULSTAR fuel assemblies have an engineered safety feature in their design to mitigate this type of accident. The safety feature consists of one-inch diameter holes located on each side of the zircaloy box just below the upper fuel pin support plate. These four bypass flow holes provide an alternate path for the coolant to enter the fuel assemblies in the unlikely event that the upper coolant inlet is blocked. The analysis shows that peak fuel centerline temperature increases only slightly during a blocked flow transient, indicating that the flow through the

[a]Maximum cladding temperature in NUREG 1537 Part1 Appendix 14.1 is listed as 1500 °C (2732 °F) for PULSTAR fuel. The cladding temperature limit of 1200 °C (2200 °F) along with a 17% cladding oxidation limit was set based on discussion with the NRC via teleconference followed up by an email on August 1, 2018.

side holes provides sufficient cooling. The blocked flow accident is analyzed in detail in Section 4.3.1.1 of this report.

2.2.4. Loss of Coolant Accident

A loss of coolant accident (LOCA) resulting in the partial or complete uncovering of the reactor core is possible since the pool liner has several large penetrations, including reactor coolant pipes and beamtubes, that would permit a drain down of the reactor pool in the event of failure. The following LOCA transients are analyzed:

- Reactor Pool Inlet Pipe Break
- Reactor Coolant Outlet Pipe Break
- Beamtube No.6 Break

In all three break scenarios, catastrophic failure is assumed. Coolant pipe break scenarios are double-ended guillotine breaks that cannot be isolated. Beamtube No.6 break scenario, the beamtube with the largest cross-sectional area, is a guillotine break on the pool side with a completely unobstructed opening into the reactor bay and is analyzed with the flapper open and with the flapper closed. The reactor is assumed to be operating at full licensed power with fission products at saturated equilibrium. Following initiation in each scenario, the break discharges primary coolant to a downstream pit or the reactor bay, the reactor trips due to a low primary coolant level scram, the primary pump stops, and the flapper valve is assumed to open or remain closed depending on the scenario. Convection heat transfer is modeled at the heat structure surfaces of the fuel rods and fuel boxes. Radial and axial heat conduction is modeled in the fuel and fuel box. Once the fuel is uncovered, radiation heat transfer is modeled from the fuel rods to the fuel box, and from the fuel box to the pool wall.

Once the core starts to uncover, the decay heat starts to raise the fuel and cladding temperatures. For the reactor coolant pipe break scenarios, the core becomes completely uncovered, while for the Beamtube No.6 break scenario the core is only partially uncovered. Temperatures of the fuel and cladding start to decrease as core heat removal by convective and radiative heat transfer exceeds core decay heat. Maximum fuel and cladding temperatures are below the safety limits.[a][3] The loss of coolant accident scenarios are discussed in detail in Section 4.3.2 of this report.

2.2.5. Reactivity Insertion Accident

Both ramp and step reactivity insertion accidents are evaluated for the PULSTAR Reactor. Both accident scenarios assume that the reactor is loaded with excess reactivity at the technical specification limit, and that other technical specification limits are at their most limiting values at initiation.

<u>Ramp Insertion Accident</u>

Several ramp insertion scenarios were evaluated, with the ramp insertion from hot zero power under forced flow conditions (i.e. a startup accident) determined to be the most limiting and bounding for all scenarios. This scenario assumes that the reactor is critical at 100 watts when the control rods are

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[a]Maximum cladding temperature in NUREG 1537 Part1 Appendix 14.1 is listed as 1500 °C (2732 °F) for PULSTAR fuel. The cladding temperature limit of 1200 °C (2200 °F) along with a 17% cladding oxidation limit was set based on discussion with the NRC via teleconference followed up by an email on August 1, 2018.

withdrawn on gang. Both 100 pcm/sec and 200 pcm/sec insertion rates were evaluated. The resulting maximum fuel and cladding temperatures remained well below the safety limits specified in NUREG 1537.[a[3] The ramp insertion accident is discussed in detail in Section 4.3.3.2 of this report.

Step Insertion Accident

The step insertion accident scenario analyzed was the fuel mishandling accident, where a fuel assembly is dropped into the critical core inserting an amount of positive reactivity equivalent to the technical specification limit for maximum fuel assembly worth (1600 pcm). The scenario assumes that the reactor is critical when the insertion happens. Three different power/flow scenarios are evaluated for the step insertion transient; hot zero power without forced flow, hot zero power with forced flow, and hot full power with forced flow. For all three power/flow conditions, the maximum fuel and cladding temperatures remained well below the safety limits specified in NUREG 1537.[a][3] The step insertion accident is discussed in detail in Section 4.3.3.3 of this report.

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[a]Maximum cladding temperature in NUREG 1537 Part1 Appendix 14.1 is listed as 1500 °C (2732 °F) for PULSTAR fuel. The cladding temperature limit of 1200 °C (2200 °F) along with a 17% cladding oxidation limit was set based on discussion with the NRC via teleconference followed up by an email on August 1, 2018.

3. RELAP Model

The thermal hydraulic analysis for the PULSTAR reactor was performed using the RELAP5/MOD3.3 code of the U.S. Nuclear Regulatory Commission (NRC). [1] RELAP5 is a light water reactor transient analysis code developed at the Idaho National Engineering Laboratory (INEL) for the U.S. Nuclear Regulatory Commission (NRC). RELAP5/MOD3.3 has been developed jointly by the NRC and a consortium of several countries and domestic organizations that were members of the International Code Assessment and Applications Program (ICAP) and its successor organization, Code Applications and Maintenance Program (CAMP). It is capable of analyzing a wide variety of thermal-hydraulic transients in nuclear and non-nuclear systems. RELAP5 is one of the most widely used system codes for analyzing reactor accidents and transients.

RELAP5/MOD3 was developed for the analysis of postulated accidents in light water reactor systems, including both large and small break loss of coolant accidents (LOCAs) as well as a full range of operational transients. The hydrodynamic model in RELAP5 is a one-dimensional, transient, two-fluid model for flow of a two-phase steam-water mixture. The non-equilibrium transient two-fluid model is represented by the conservation equations of mass, momentum, and energy for each phase. The steam phase can contain non-condensable components and the water phase can have a solute component. Special process models are available to handle choked flow, abrupt area changes, and counter-current flow.

Solid components are modeled by heat structures with internal heat generation. Heat transfer within the structures is by one-dimensional heat conduction. A full boiling curve is implemented in the code for modeling heat transfer between heat structures and the coolant. Reactor power and decay power are calculated by a point kinetics model with reactivity feedback. In RELAP5 a hydraulic system is constructed by connecting fluid components, such as pipes, valves, pumps, etc., in series or in parallel. Geometric data and the initial thermodynamic state of the fluid are required for the interconnecting components. The initial flow rate is required at the junctions between two components. Heat structures are defined with the heat transfer surface facing the coolant in a hydraulic component. Time varying boundary conditions can be specified in terms of fluid flow rate or the thermodynamic state of the fluid. Control system components are available in RELAP5 to model system dynamic behavior such as component trips and the evaluation of system variables.

The RELAP5 version used for the PULSTAR analysis is US-NRC RELAP5/MOD3.3 Patch04. The QA installation verification was carried out on the Windows platform using two sample problems: *Edward Pipe* and *Zion 1 SBLOCA*. The verification run showed that the results coincide with those of the software supplier. Therefore, it is confirmed that RELAP5 was correctly installed and configured.

3.1. Modeling of the PULSTAR Reactor

The RELAP5 model of the PULSTAR facility is shown in Figure 3-1 through Figure 3-3. The model was developed based on PULSTAR drawings and onsite walk downs. The modeling approaches for the reactor core, coolant systems, and reactor pool are summarized below.

The RELAP5/MOD3.3 model of the PULSTAR simulates the transport of heat and coolant in the primary system. The pool and the primary coolant loop are represented by a series of hydrodynamic volumes. Fuel assemblies in the core region are represented by heat structures. Fission and decay power are calculated by using the point kinetics model in RELAP5. Schematic diagrams showing the main

components of the PULSTAR primary system are shown in Figure 3-1 through Figure 3-3, each representing the reactor pool and core configuration and the coolant loop configuration. The discussion of the PULSTAR model will be grouped into four sub-sections: the reactor pool, the reactor core, the primary coolant loop, and the secondary coolant loop. A component number, as defined in the RELAP5 input deck, is used to identify each hydrodynamic volume modeled.

3.1.1. Reactor Pool

The reactor pool is divided into a number of interconnected hydrodynamic volumes as shown in Figure 3-1. During normal operation, primary coolant heated from the reactor flows downward to the reactor coolant loop, is cooled by the heat exchanger and then returns to the bottom of the reactor pool. The top of the reactor pool is open to atmosphere, which is modeled as a time-dependent volume in RELAP5. The flapper valve at the core outlet plenum is modeled to open by loss of differential pressure between the pool and the plenum when primary coolant flow stops. Under certain transient conditions the flapper is modeled both open and closed.

3.1.2. Reactor Core

Figure 3-2 shows the fuel channel modeling and nodalization in the reactor core. The core consisting of 5×5 fuel boxes each containing 5×5 fuel rods is modeled with four lumped fuel channels. A total of 25 fuel channels are lumped into three averaged fuel channels, each representing 5, 9 and 10 fuel channels, and one hot channel that represents a highest power fuel channel. Axial nodalization of flow channels consists of 14 volumes, with 10 volumes in the active core and additional volumes at the top and bottom of the assemblies. Fuel channels are modeled open to the reactor pool via the top inlet and the side flow holes at the top of the active core. Thus, core inlet flow is formed in two flow paths, one from the core top and one from the side flow holes. Downward core flow is collected in the core outlet plenum and then flows to the primary coolant loop.

Fuel rods in a lumped fuel channel are modeled with heat structures. A single lumped fuel pin is used to represent fuel rods in a lumped fuel channel and an additional hottest pin is modeled in a hot channel for assessment of safety margin. Fuel material properties are obtained from IAEA-TECDOC-1496. [6] One-dimensional radial or transverse conduction in a heat structure is modeled and additional axial heat conduction is modeled when the heat structures are uncovered. When the surface of the heat structure is uncovered, radiation heat transfer turns on from fuel pin to fuel box and then to the heat sink representing the pool wall. Convective heat transfer is modeled at surfaces of fuel pins and fuel boxes. An AECL critical heat flux look-up table is used to estimate the fuel DNBR transients.

3.1.3. Primary Coolant Loop

The primary coolant flow path in the PULSTAR is a single loop with components located in series. Figure 3-3 depicts the layout of the primary coolant loop from the reactor outlet to the reactor pool inlet.

The primary coolant loop is modeled with primary components, ^{16}N delay tank, coolant pump, heat exchanger and pipes connecting the components. Piping layouts and elevations are modeled as currently installed in the PULSTAR facility. For simplicity, the heat exchanger has been modeled as a tube and shell to meet the energy balance. This simplification does not have a significant effect on the RELAP5 analysis.

3.1.4. Secondary Coolant Loop

The secondary cooling loop is modeled simply as a once through circuit. At one end, a source supplies the cooling water to the heat exchanger. The other end flows to a sink (cooling tower).

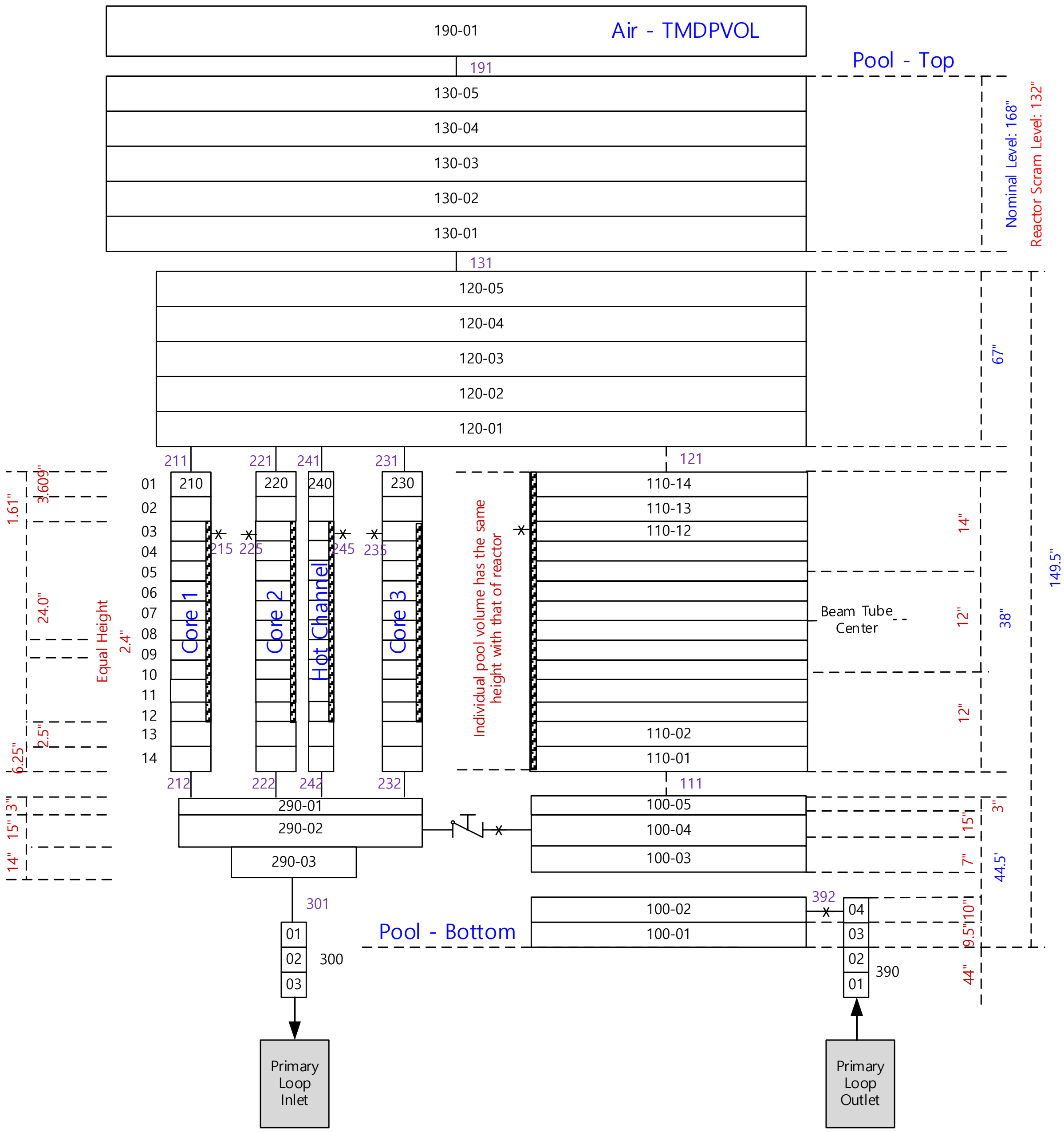


Figure 3-1 – RELAP5 Model Nodalization of the PULSTAR Reactor Pool and Core

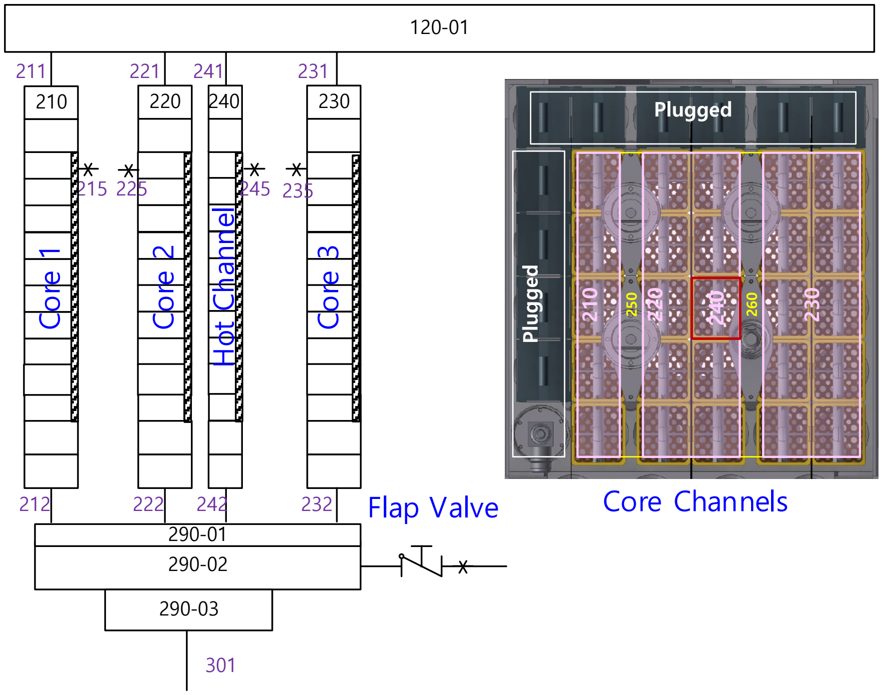


Figure 3-2– PULSTAR Fuel Channel Modeling and Nodalization

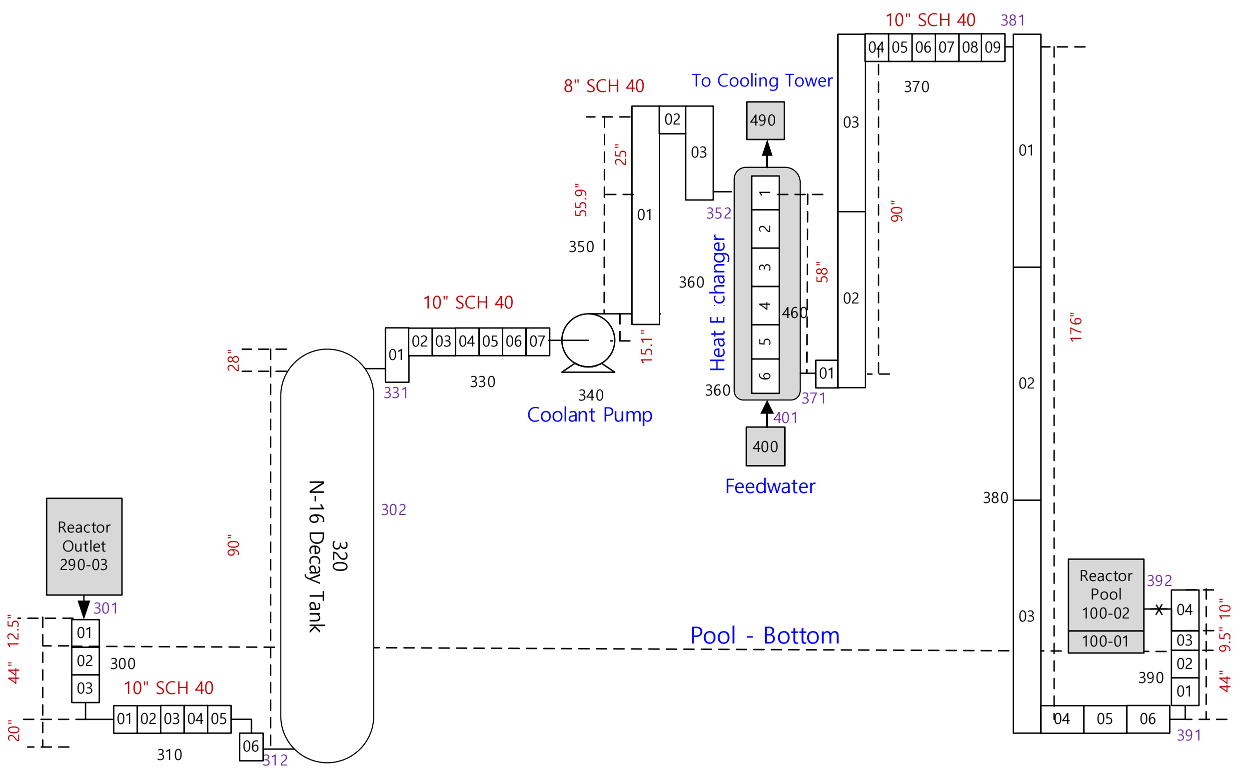


Figure 3-3– RELAP5 Model Nodalization of the PULSTAR Coolant System

3.2. Point Kinetics Input Data

The RELAP5 reactor kinetics model calculates the total reactor power as the sum of fission power and fission product decay power. Fission power is calculated from the point kinetics model with delayed neutron precursor groups with an effective delayed neutron fraction of 0.00733. [2] The prompt neutron generation time is conservatively assumed to be 36 µsec. [8] The option selected to calculate decay heat in RELAP5 is to use the ANSI/ANS 5.1-1979, Decay Heat Power in Light Water Reactors [7] multiplied by 1.0. The total reactivity of the gang rod is 3.457 dollars. The scram motion of the control rods is assumed to follow standard particle dynamic models for dropped objects. It is assumed that rod drop times from full out to full in will be at the technical specification limit of 1.0 seconds. A conservative time delay has been incorporated in the start of control rod motion after the initiation of a scram. It is assumed that starting from any position there will be 0.05 second delay. [8]

3.2.1. Core Flow Distribution

Flow through the reactor core is calculated based on the number of fuel assemblies that have been lumped into each core channel.

3.2.2. Core Power Distribution

The PULSTAR core has 25 fuel assemblies. Each fuel assembly contains 25 fuel pins. It is impractical to model individually all 625 fuel pins, instead the fuel assemblies are lumped into four groups. Each group of fuel assemblies is represented as an idealized core channel. The core channel flow paths are modeled as connected in parallel between inlet from the reactor pool and core outlet plenum. Power distribution to each channel is obtained by lumping power distributions from NCSU PULSTAR core physics analysis. [2] Conservatively, a core peaking factor of 3.0 is used.

3.2.2.1. Hot Channel

The hot channel is selected from the NCSU PULSTAR core analysis. Active fuel is axially divided into 10 nodes and the hottest pin in the hot channel is conservatively assumed to have a peaking factor equal to 3.0.

3.3. Reactor Trips

The initiation of safety systems is defined in RELAP5 in the form of trip variables. Each trip is identified by a number. A trip or scram can be initiated by a number of conditions in the reactor. The trips that have been modeled for the analysis of the PULSTAR accidents are the power, flow and pool level trips. Table 3-1 lists the scram values and the corresponding time delays as assumed in the RELAP5 model.

Table 3-1 – SCRAM Set-point and Time Delays

	Low Primary Flow*	High Power		Pool Level above Core
		Forced Flow	Zero Flow	
Scram Setpoint	900 gpm	2.0 MWt	0.25 MWt	204 inches
Time Delay	0.05 sec	0.05 sec	0.05 sec	0.05

* In case of loss of flow accidents, additional time delay of 2 sec is considered

4. 2 MWt Thermal-Hydraulic Analysis

4.1. Steady-State Analysis

4.1.1. Forced Convection Mode

Table 4-1 provides a summary of the steady-state initialization under limiting conditions of operation (LCO), which are the limiting safety system settings (LSSS), compared with reference nominal operating conditions. Nominal values listed in Table 4-1 represent typical operating conditions at full power and are provided for comparison to simulated values at the LSSS. Calculations were performed at the simulated limiting conditions, not at the nominal values. The resulting thermal margins for steady-state forced flow analyzed at the limiting conditions are summarized in Table 4-2.

Figure 4-1 shows the fuel temperature profiles for limiting conditions at a power level of 2.0 MWt. The minimum DNBR remains well above the safety limit of 2.0. There is sufficient subcooled temperature margin in the hot fuel channel exit to preclude bulk boiling. Fuel centerline and cladding temperatures are well below temperature limits. These calculations assume a hot pin power peaking factor of 3.0.

Table 4-1 – Steady-State Initialization for Forced Convection

Parameter	**Nominal Conditions**	**Simulated Limiting Conditions**
Reactor Power (MWt)	1.80	2.0
Power Peaking Factor	2.54	3.0
Primary Flow (gpm)	1000	900
Core Inlet Temperature (°F)	105	117
Core Temperature Rise (°F)	12.4	15.3
Pool Water Level above Core (inches)	240	204
Secondary Flow (gpm)	1000	1000
Secondary Inlet Temperature (°F)	92	99
Secondary Pressure (psig)	18.5	18.5

Table 4-2 – Steady-State Thermal Margins at Limiting Operating Conditions for Forced Convection

Parameter	Value
Peak Fuel Centerline Temperature (°F)	1110.8
Peak Cladding Temperature (°F)	226.9
Minimum DNBR	> 19

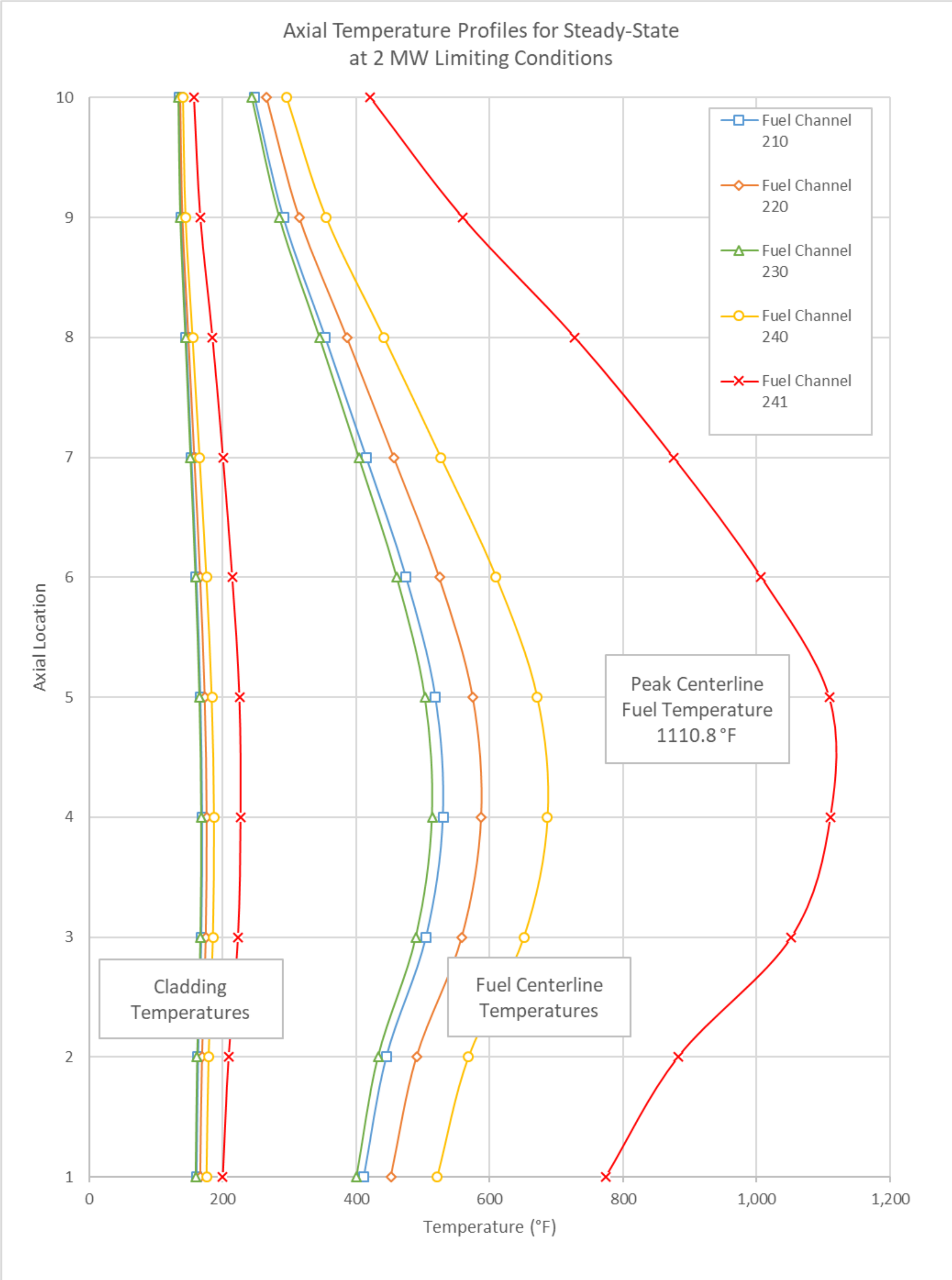


Figure 4-1– Steady-State Axial Fuel Temperature Profiles – Limiting Conditions for Forced Convection

4.1.2. Natural Circulation Mode

For natural convection mode, the analysis used the RELAP5 Model to evaluate various power levels with no forced flow and the flapper open to determine when the major safety parameters were challenged.

Table 4-3 provides a summary of the steady-state initialization under limiting conditions of operation (LCO), which are the limiting safety system settings (LSSS), compared with reference nominal operating conditions. Nominal values listed in Table 4-3 represent typical operating conditions for natural circulation mode and are provided for comparison to simulated values at the LSSS. Calculations were performed at the simulated limiting conditions, not at the nominal values. The resulting thermal margins for steady-state natural circulation analyzed at the limiting conditions are summarized in Table 4-4. Figure 4-2 shows the hot pin fuel temperatures and DNBR under natural circulation for limiting conditions at a power level of 250 kWt. The minimum DNBR remains well above the safety limit of 2.0. There is sufficient subcooled temperature margin in the hot fuel channel exit to preclude bulk boiling. Fuel centerline and cladding temperatures are well below temperature limits. These calculations assume a hot pin power peaking factor of 3.0.

Table 4-3 – Steady-State Initialization for Natural Circulation

Parameter	Nominal Conditions	Simulated Limiting Conditions
Reactor Power (kWt)	100	250
Power Peaking Factor	2.54	3.0
Primary Flow (gpm)	0	0
Core Inlet Temperature (°F)	105	120
Pool Water Level above Core (inches)	240	168

Table 4-4 – Steady-State Thermal Margins at Limiting Operating Conditions for Natural Circulation

Parameter	Value
Peak Fuel Centerline Temperature (°F)	343.9
Peak Cladding Temperature (°F)	242.9
Minimum DNBR	22.8

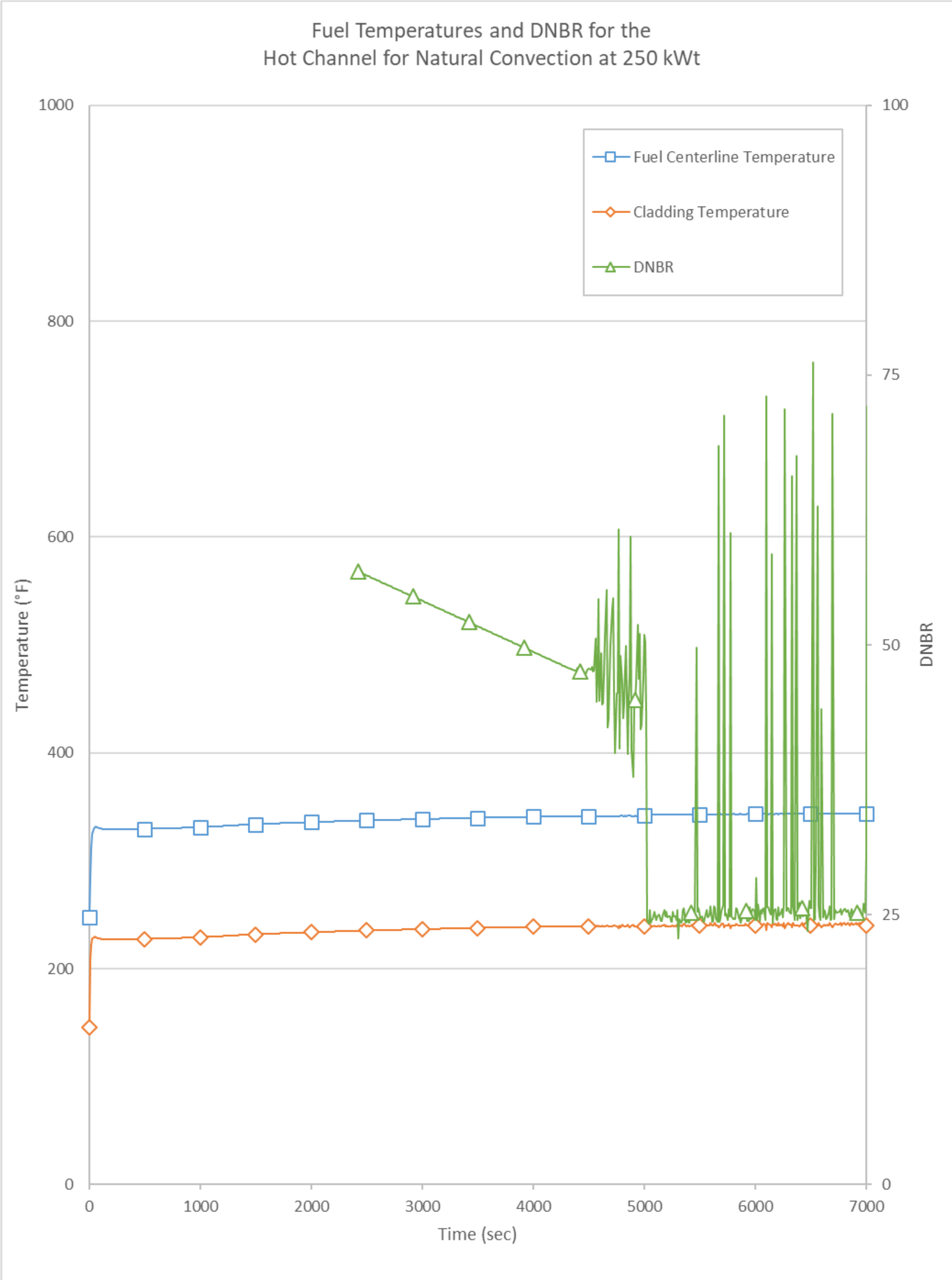


Figure 4-2 – Hot Pin Temperatures and DNBR for Natural Circulation at 0.25 MWt

4.2. Safety Limit and Limiting Safety System Setting Determination

The major safety parameters were evaluated for steady-state operation for both forced convection and natural circulation mode of operation to determine the limit at which one can be assured that the integrity of the fuel and cladding will be maintained.

4.2.1. Forced Convection Model

For forced convection mode, the analysis used the RELAP5 Model to evaluate various power vs. flow ranges to determine when the major safety parameters were challenged, where 1000 gpm is equal to 100% flow.

The initial conditions were set to the limiting conditions:

- Pool level reduced from 240 inches to the Safety Limit of 168 inches from the top of the core.
- Pool temperature increased from nominal operating temperature of 105.0 °F to the Safety Limit 120.0 °F.
- Flow rate was set to 50% of nominal full core flow (i.e. 500 gpm).

Power levels ranging from 0.5 MWt, to 5.0 MWt were evaluated to see if the major safety parameters were challenged. For each power level the flow rate was reduced in 5% increments from nominal 50% full flow until the major safety parameters were challenged. Table 4-5 summarizes the results of the analysis for the hot channel and the flow rates when the major safety parameters are challenged.

Table 4-5 – Thermal Margins for Forced Convection at Various Power Levels

Core Power Level (MWt)	Onset of Flow Instability (gpm)	Core Bulk Boiling (gpm)	DNBR < 2.0 (gpm)	Fuel Failure (gpm)
0.5	75	75	Does not Occur	Does not Occur
1.0	125	125	0	0
2.0	200	200	200	50
3.0	275	275	275	275
4.0	400	400	400	400
4.5	450	450	450	450
5.0	500	500	500	500

The Safety Limit is a value of the chosen variable (e.g., power level, primary flow rate and pool level) at which there is sufficient margin that the design criteria will not be exceeded. The interrelated power

levels and primary flow rates required to exceed the design criteria in the hot channel are depicted in Figure 4-3 and can be used to establish a Safety Limit curve. Even though the NCSU PULSTAR operates at only one forced convection flow condition, the Safety Limit can be established as a function of power and flow and is shown in Figure 4-4.

A limiting safety system setting value is selected with sufficient margin so that a Safety Limit will not be exceeded (see Table 4-6). The margin between this value and the safety limit should be sufficient to allow for corrective action by the safety system to return the situation to normal or to shut the reactor down before the safety limit would be reached. This means that circuit response times, transient characteristics, and measurement uncertainties must be taken into account.

Table 4-6– Safety Limits and Limiting Safety System Settings for Forced Convection Operation

Parameter		**Safety Limit**	**Limiting Safety System Settings**
P	Power (MW)	The combination of true values of reactor thermal power (P) and reactor coolant flow rate (W) shall not exceed the limits shown in Figure 4-4 under any operating conditions. The limits are considered exceeded if the point defined by the true values P and W is at any time outside the operating envelope shown in Figure 4-4	2.0
W	Flow (gpm)		900
H	Pool Level (ft)	The true value of the pool water level (H) shall not be less than 14 feet above the top of the core	17
T_{inlet}	Bulk Pool Temperature (°F)	The true value of the reactor coolant inlet temperature (bulk pool temperature) shall not be greater than 120 °F	117

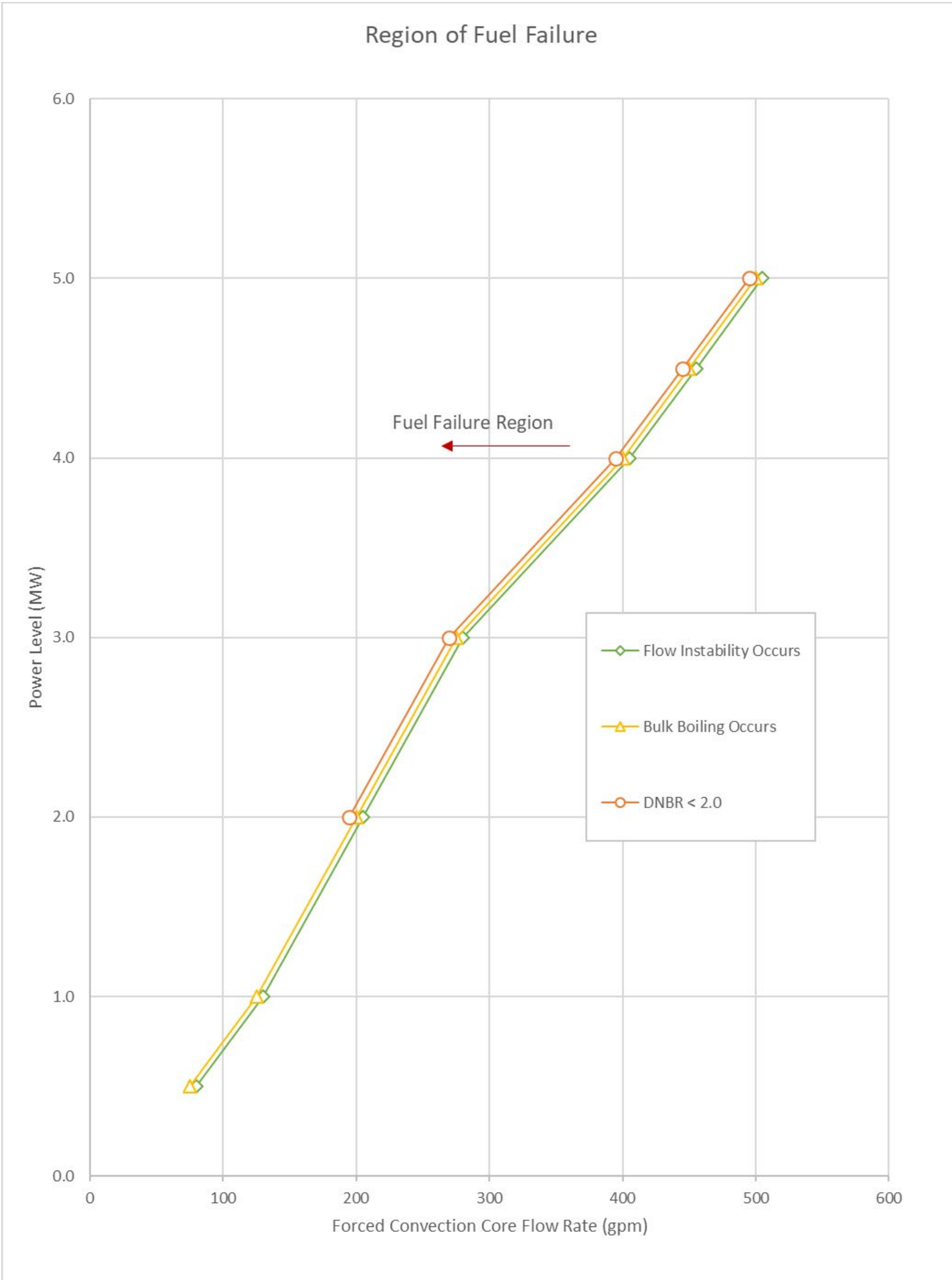


Figure 4-3 – Power Level and Flow Rates Required to Challenge Major Safety Parameters

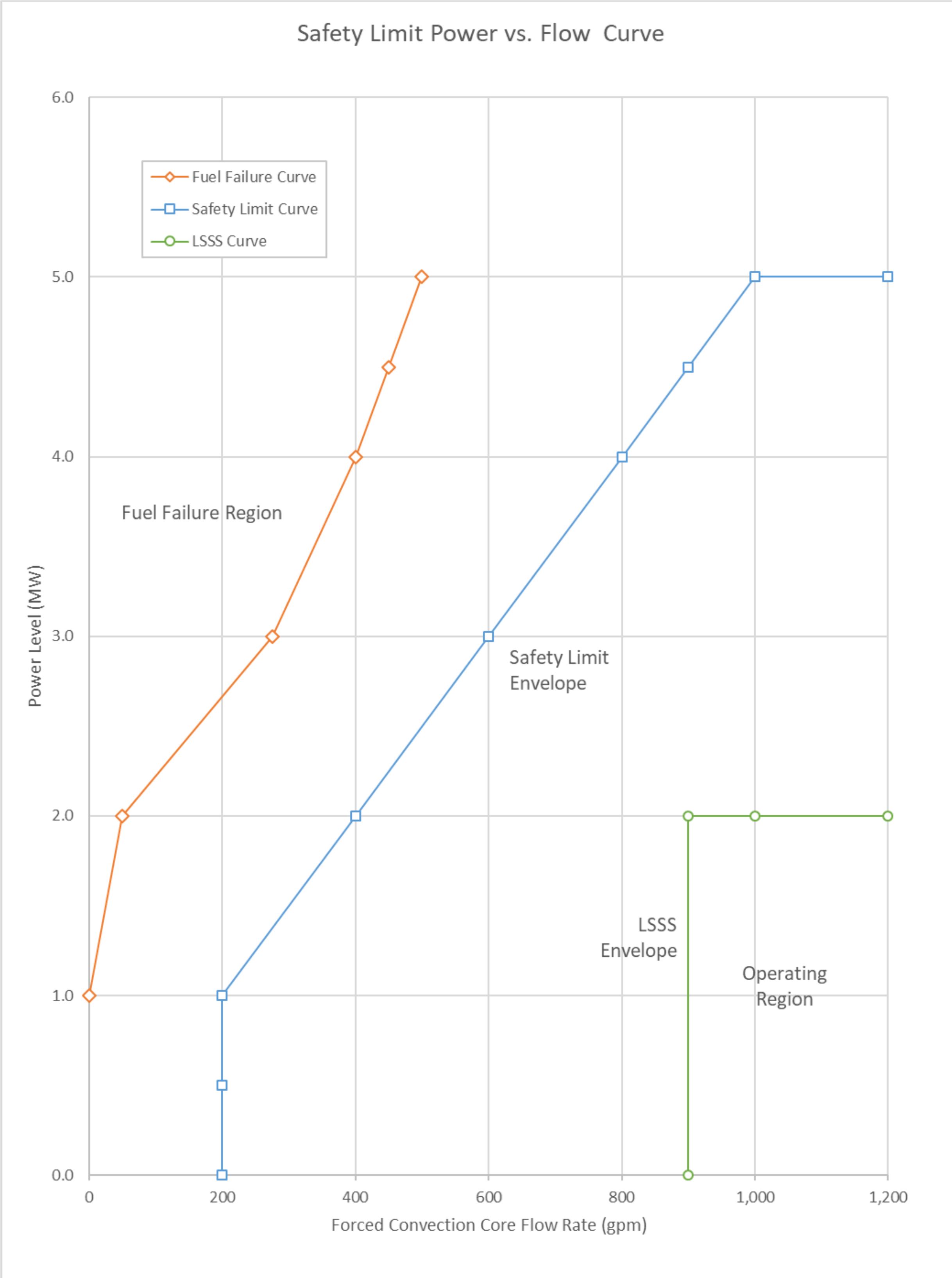


Figure 4-4– Safety Limit Power vs Flow Curve

4.2.2. Natural Circulation Mode

For natural circulation mode the analysis used the RELAP5 Model to evaluate various power levels with no forced flow and the flapper open to determine when the major safety parameters were challenged.

The initial conditions were set to the limiting conditions:

- Pool level reduced from 240 inches to the Safety Limit of 168 inches from the top of the core.
- Pool temperature increased from nominal operating temperature of 105 °F to the Safety Limit 120 °F.
- Forced flow rate was set to zero.
- Flapper valve was set open and natural circulation was analyzed through the core for two hours at various power levels.

Power levels ranging from 100 kWt to 1.6 MWt were evaluated to see if the major safety parameters were challenged. The major safety parameters [3,4] evaluated were:

- Core Boiling
- Flow Instability
- DNBR
- Fuel Centerline Temperatures
- Fuel Cladding Temperatures[a]

There is no core boiling or flow instability for natural circulation at 100 kWt. Since the only heat sink is the reactor pool, pool temperature continuously increases with time, which results in localized core boiling at power levels greater than 100 kWt. The higher the core power the earlier the onset of core boiling. With onset of localized core boiling, single phase natural circulation transitions to two phase natural circulation and flow instability occurs in core channels.

For reactor power levels of 1.0 MWt and greater, hot pin DNBR decreases below 2.0 but the core is satisfactorily cooled by oscillatory two phase flow followed by core bulk boiling and fuel integrity is assured for power levels up to 1.4 MWt (see Figure 4-5 and Figure 4-6, and Table 4-7). For the case of 1.6 MWt there is significant core voiding which drastically reduces heat transfer from the fuel (i.e. DNBR << 1.0) resulting in fuel failure (see Figure 4-7).

The power levels required to reach the major safety parameters in the hot channel are listed in Table 4-7 and can be used to establish the safety limit. Fuel and cladding temperatures are shown in the figures below. The power level of 1.0 MWt was chosen as the safety limit because it satisfied the safety parameters for fuel and cladding temperatures, assuring that the fuel and cladding integrity will be maintained. Figure 4-8 shows that maximum fuel and cladding temperatures for the various power levels along with the temperature limits.

Under natural convection conditions, bulk boiling can occur at power levels greater than 250 kWt resulting in undesirable releases of ^{16}N activity following possible bubble rise in the pool water. To eliminate this

[a]Maximum cladding temperature in NUREG 1537 Part1 Appendix 14.1 is listed as 1500 °C (2732 °F) for PULSTAR fuel. The cladding temperature limit of 1200 °C (2200 °F) along with a 17% cladding oxidation limit was set based on discussion with the NRC via teleconference followed up by an email on August 1, 2018.

possibility, the limiting safety system setting given is not related to fuel damage but to this potential for ^{16}N activity at the pool top. Safety Limits and Limiting Safety System Settings for natural circulation operation are listed in Table 4-8.

Table 4-7 – Thermal Margins for Natural Circulation at Various Power Levels

Core Power Level MWt	Onset of Local Boiling and Flow Instability (sec)	Onset of Core Bulk Boiling (sec)	Hot Pin Fuel Centerline Temperature (°F)	Hot Pin Cladding Temperature (°F)	Minimum DNBR
0.1	Does not Occur	Does not Occur	243.5	203.0	Not Calculated
0.25	~4500	Does not Occur	343.9	242.9	22.8
0.5	~1800	~6900	456.9	248.0	9.8
1.0	Immediately	~3000	734.1	342.7	1.1
1.4	Immediately	~2300	983.1	424.2	0.7
1.6	Immediately	~2000	3553.8	2603.3	0.1

Table 4-8– Safety Limits and Limiting Safety System Settings for Natural Circulation Operation

Parameter		Safety Limit	Limiting Safety System Settings
P	Power (MW)	1.0	0.25
H	Pool Level (ft)	The true value of the pool water level (H) shall not be less than 14 feet above the top of the core	17
T_{inlet}	Bulk Pool Temperature (°F)	The true value of the reactor coolant inlet temperature (bulk pool temperature) shall not be greater than 120 °F	117

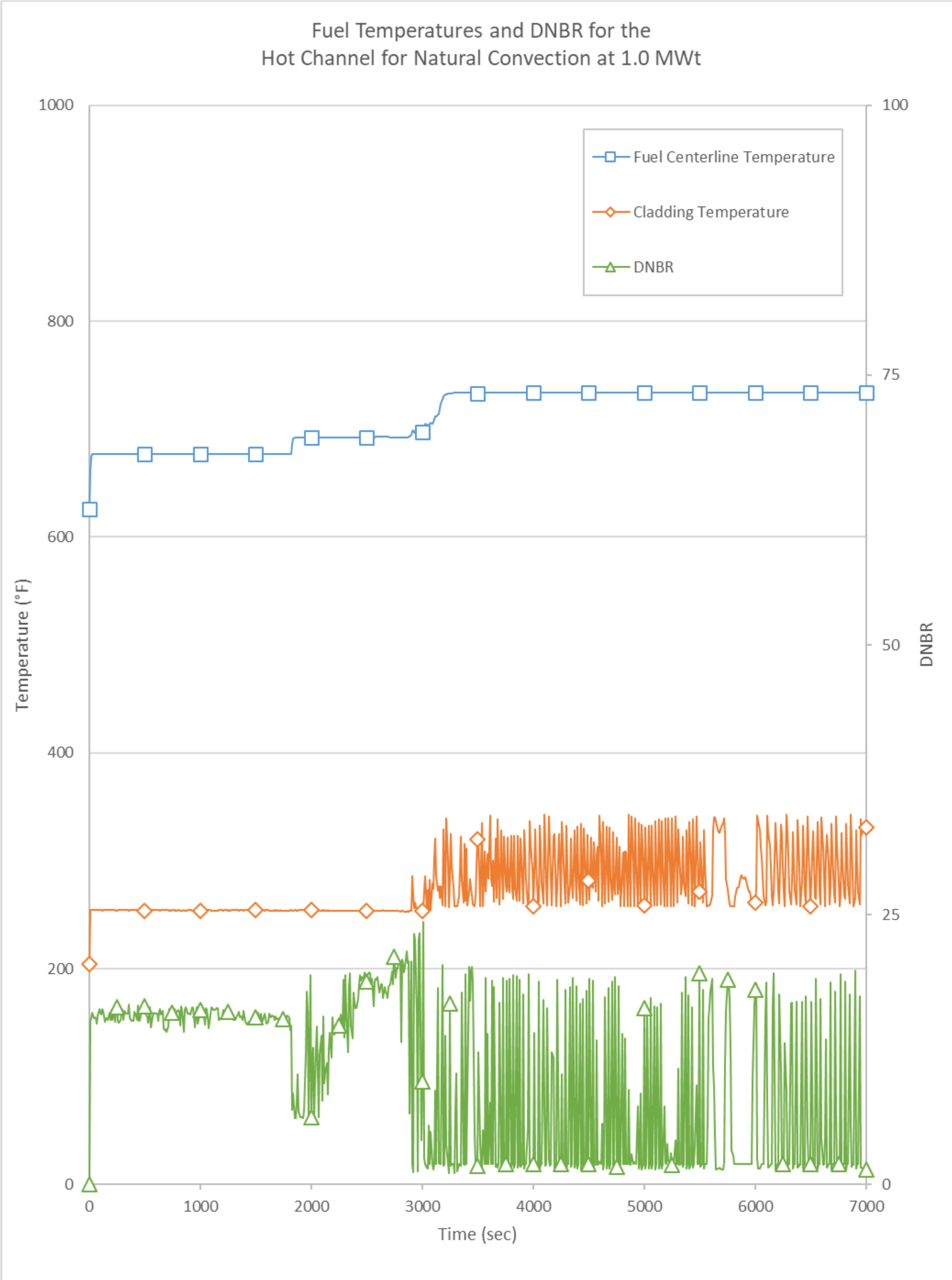


Figure 4-5 – Hot Pin Temperatures and DNBR for Natural Circulation at 1.0 MWt

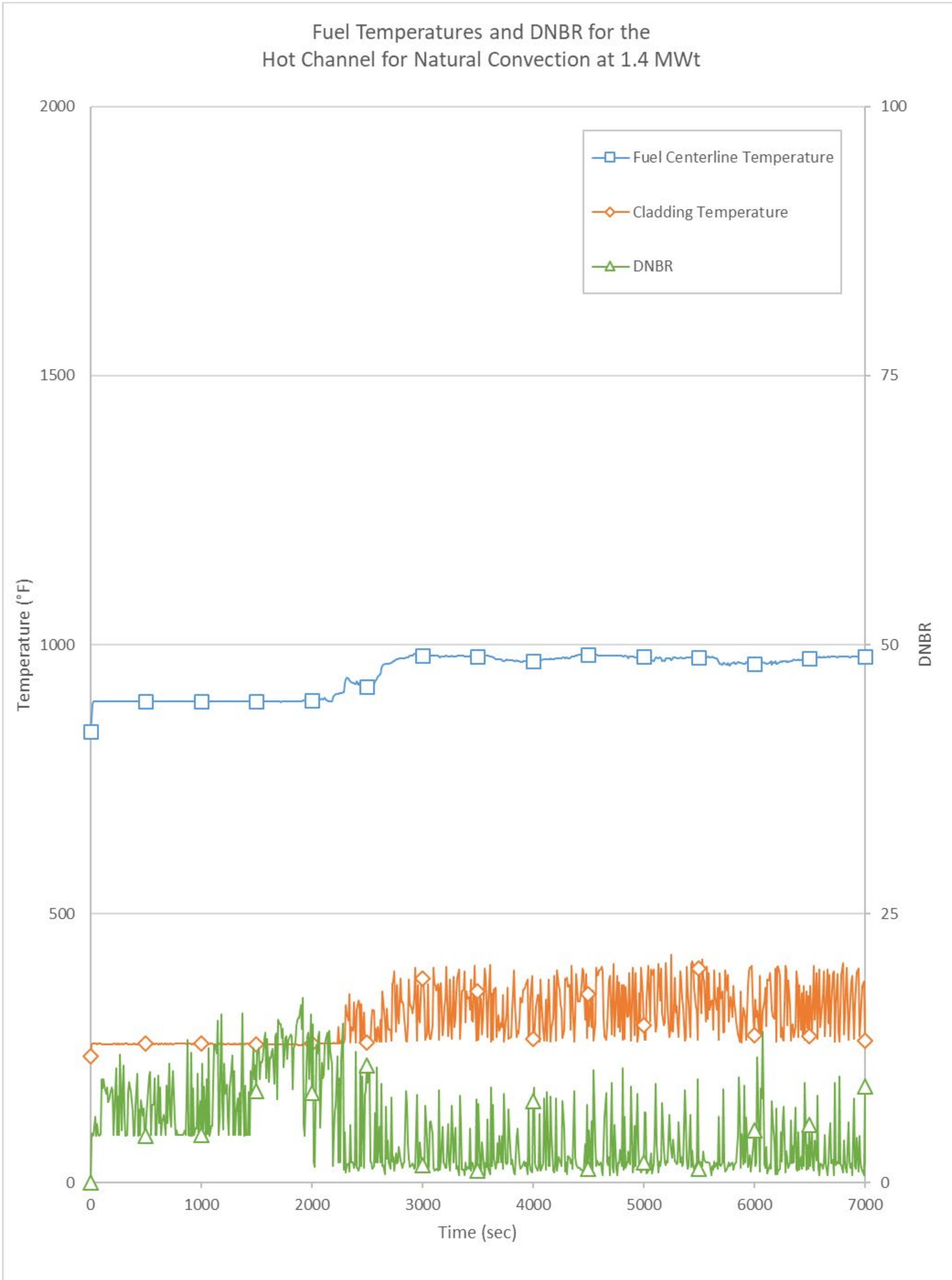


Figure 4-6 – Hot Pin Temperatures and DNBR for Natural Circulation at 1.4 MWt

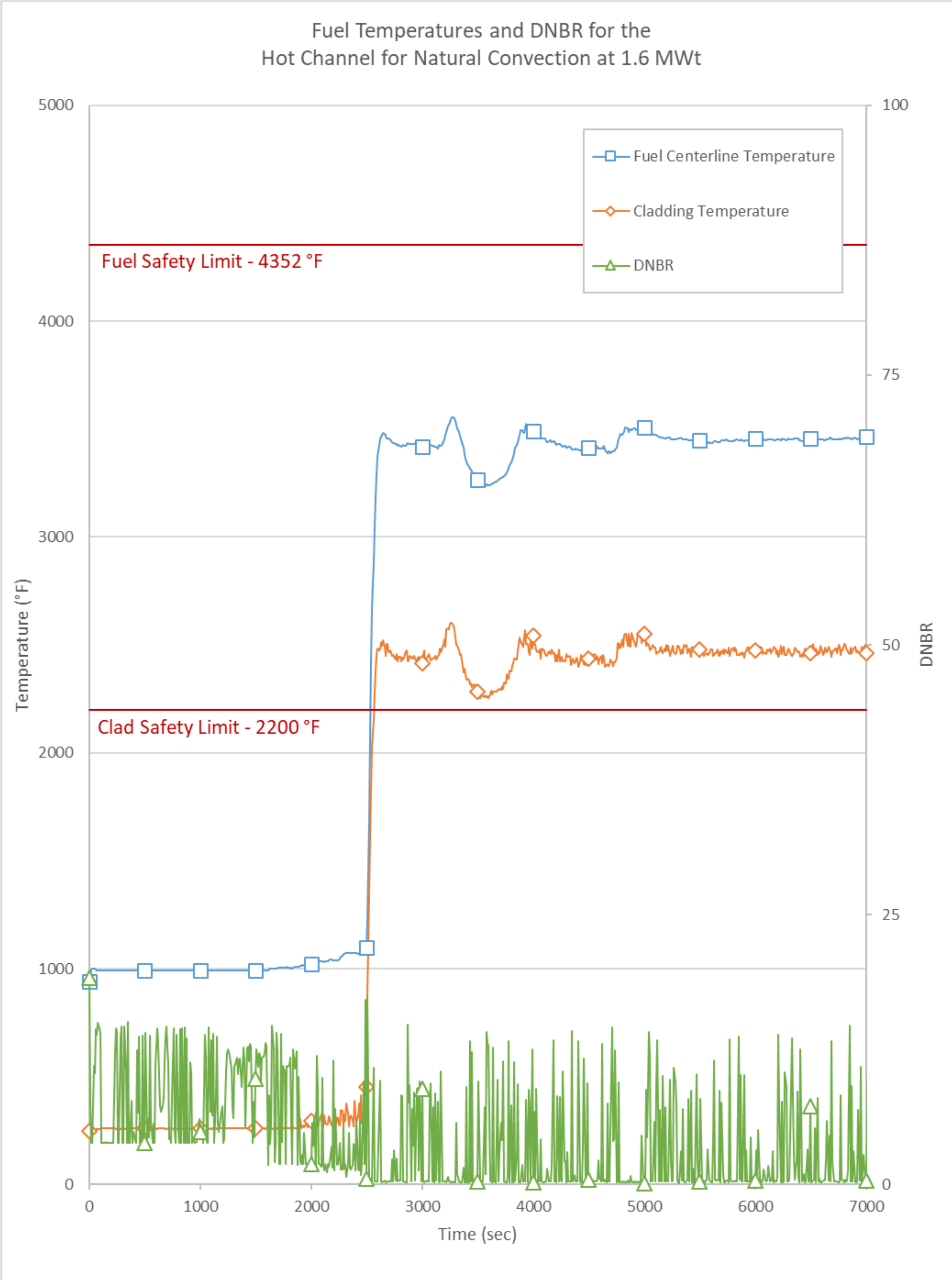


Figure 4-7 – Hot Pin Temperatures and DNBR for Natural Circulation at 1.6 MWt

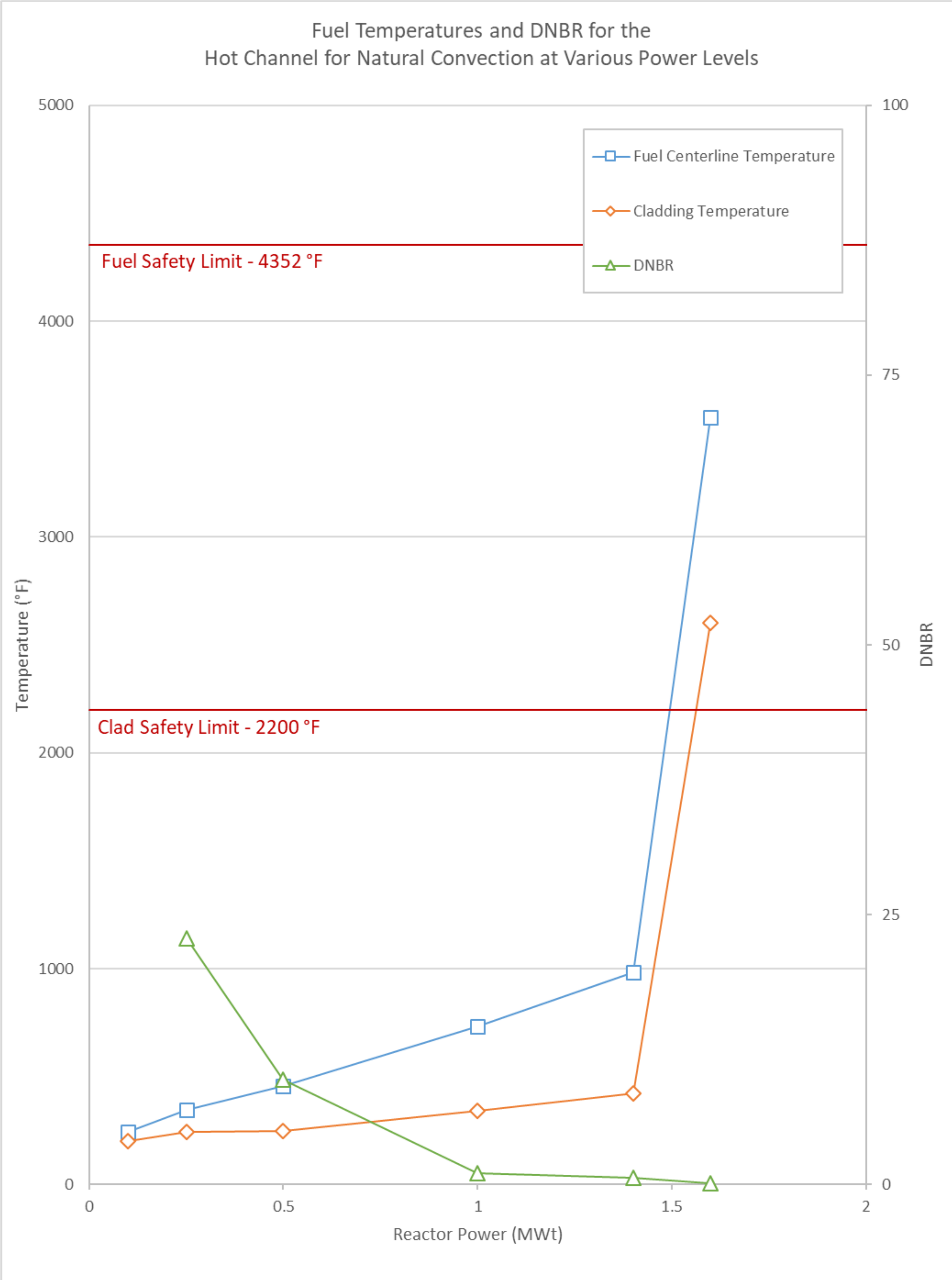


Figure 4-8 – Fuel Temperatures and DNBR for Natural Circulation

4.3. 2 MWt Accident Analysis

This report provides assessments for the following accidents:

- Loss of Primary Flow
- Loss of Coolant
- Reactivity Insertion

For all postulated accidents, the reactor is assumed to scram when a Limiting Safety System Setting (LSSS) signal is received by the Reactor Safety System (RSS).

4.3.1. Loss of Flow Accident

This accident is one of the most limiting for forced convection cooled non-power reactors, where the forced flow is downward through the reactor core. Upon loss of forced downward coolant flow through the core, coolant flow in the core must reverse to upward natural convection cooling. During the flow reversal, heat transfer may be inadequate in the core which may challenge major safety parameters and ultimately the integrity of the fuel cladding. Loss of coolant flow may also occur if a foreign object obstructs a coolant flow path. Some initiators of loss of coolant flow accidents are the following:

- Loss of electrical power
- Failure of a pump or other component in the primary coolant system
- Blocking or significant decrease in flow in one or more fuel coolant channels

4.3.1.1. Blocked Flow Accident

The PULSTAR fuel assembly has an engineered safety feature in its design to eliminate the blocked flow type of accident. This safety feature consists of 4 one-inch diameter holes in the zircaloy box just below the upper fuel pin spacer grid located in each fuel assembly box. These holes have approximately the same total flow area as the upper coolant inlet of the fuel assembly. Since all 25 fuel pins in a fuel assembly have a common flow channel, the four holes will provide a path for the coolant in the unlikely event that a foreign object (e.g., gaskets, plastic sheets, etc.) falls over the upper coolant inlet of the fuel assembly.

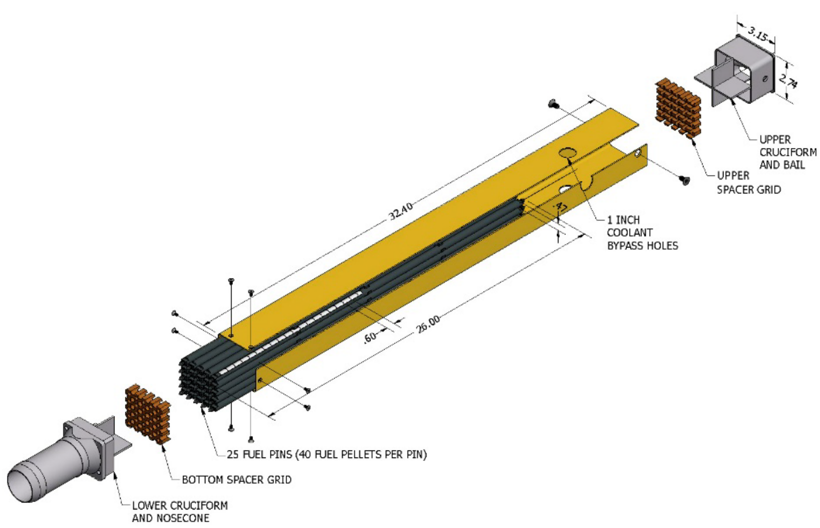


Figure 4-9 – Expanded View of a NCSU PULSTAR Fuel Assembly Showing the 1-inch Bypass Flow Holes

When local flow blockage at the top of the core inlet happens, main coolant flow to the blocked fuel channel is redirected from the fuel channel inlet at top to the side bypass flow holes. Since there is no monitoring that can detect flow blockage, proper operator actions may not take place and the reactor may erroneously continue operating at full power. Therefore, it is important to show that core safety parameters are not challenged in the case of core flow blockage.

To demonstrate core safety, a complete blockage of the hot fuel assembly at the top is conservatively assumed for analysis. The reactor is at full power operation at the limiting conditions and neither reactor scram nor operator actions are credited. Table 4-9 summaries the initial conditions for a blocked flow accident. The simulation is run until a stable operating condition is achieved provided by the redirected flow through the side flow holes.

Figure 4-10 shows flows through the hot fuel channel exit and through the side bypass holes resulting from a complete flow blockage at the top of the fuel assembly. Table 4-10 summarizes the flow distribution in the hot fuel channel in normal and blocked conditions. Hot channel exit flow decreases from 4.95 lbm/sec to a stable 4.13 lbm/sec for the blocked condition, where there is zero flow from the channel top and main flow is redirected through the side bypass flow holes. Flow through the side bypass flow holes increases from 2.08 lbm/sec to 4.13 lbm/sec, which becomes the sole contributor to fuel cooling in the blocked channel. Peak fuel centerline temperature increases slightly from 1110.8 °F to 1124.2 °F (see Figure 4-11), which implies that the flow through the side bypass flow holes provides sufficient fuel cooling. As shown in Table 4-11, there is no noticeable change in the overall reactor system parameters. From the analysis, it can be concluded that core safety is ensured by the flow through the side bypass flow holes in the event of a complete blockage of a fuel assembly.

Table 4-9 – Initialization – Blocked Flow Accident

Parameter	Nominal	Simulated Limiting Conditions
Reactor Power (MWt)	1.8	2.0
Power Peaking Factor	2.54	3.0
Primary Coolant Flow Rate (gpm)	1000	900
Pool Temperature (°F)	105	117
Core Temperature Rise (°F)	12.4	15.3
Secondary Coolant Flow Rate (gpm)	1000	1000
Secondary Coolant Temperature (°F)	92	98.82

Secondary Coolant Pressure (psig)	18.5	18.5
Pool Level from Top of Core (inches)	240	204
Peak Fuel Centerline Temperature (°F)		1110.8
Peaking Cladding Temperature (°F)		226.9

Table 4-10 – Hot Channel Parameters

Parameters	Unblocked Fuel Assembly	Blocked Fuel Assembly
Hot Channel Inlet Flow (lbm/sec)	2.87	0.0
Hot Channel Flow Hole Bypass Flow (lbm/sec)	2.08	4.13
Hot Channel Exit Flow (lbm/sec)	4.95	4.13
Hot Channel Exit Coolant Temperature (°F)	137.5	141.5
Peak Fuel Centerline Temperature (°F)	1110.8	1124.2
Peak Fuel Cladding Temperature (°F)	226.9	239.9

Table 4-11 – System Parameters

Parameters	Unblocked Fuel Assembly	Blocked Fuel Assembly
Reactor Power (MWt)	2.0	2.0
Core Inlet Temperature (°F)	117.0	117.0
Core Outlet Temperature (°F)	132.4	132.4
Loop Flow (lbm/sec)	123.6	123.5

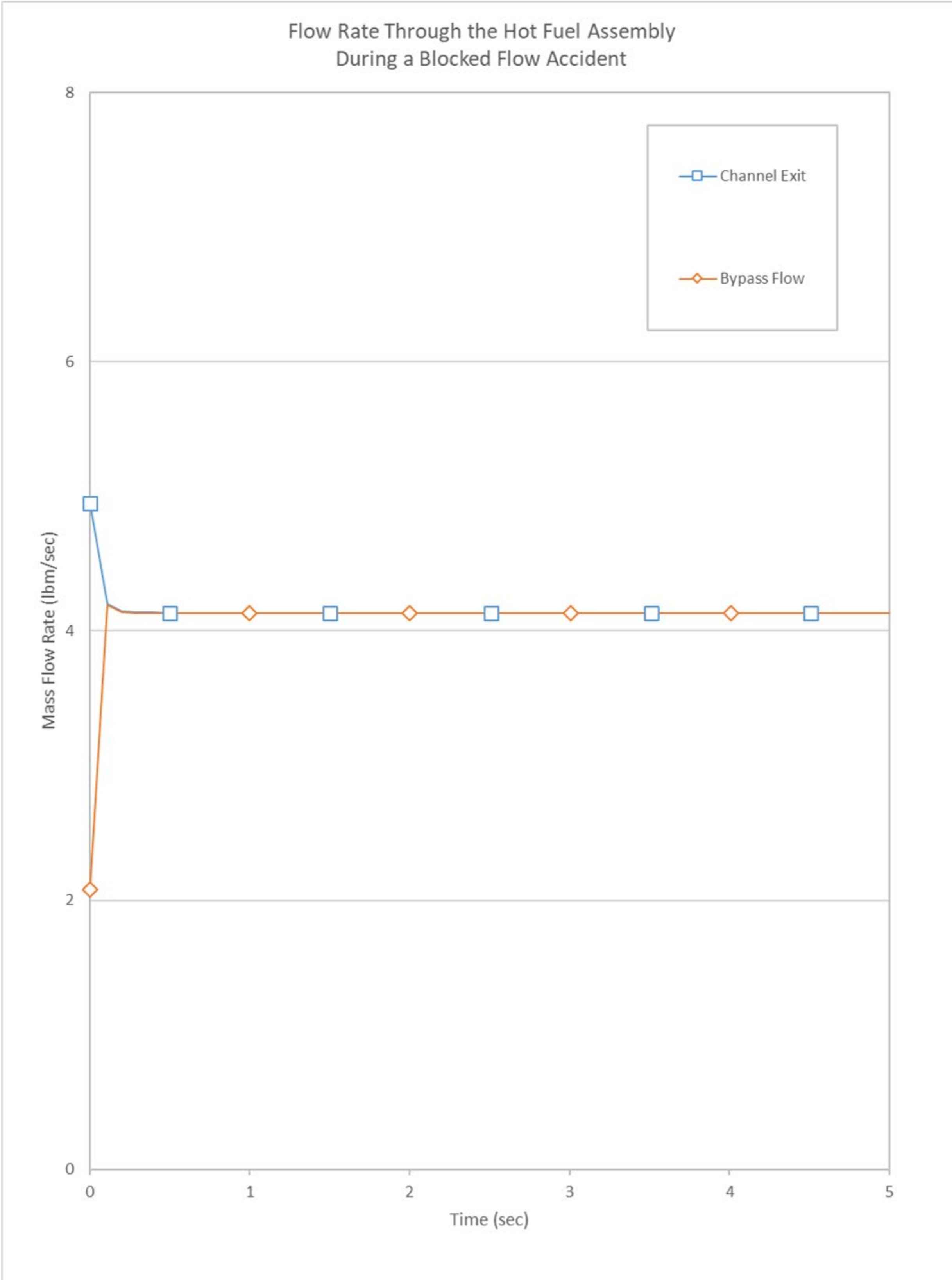


Figure 4-10 – Hot Channel Fuel Assembly Flow Rates for a Blocked Flow Accident

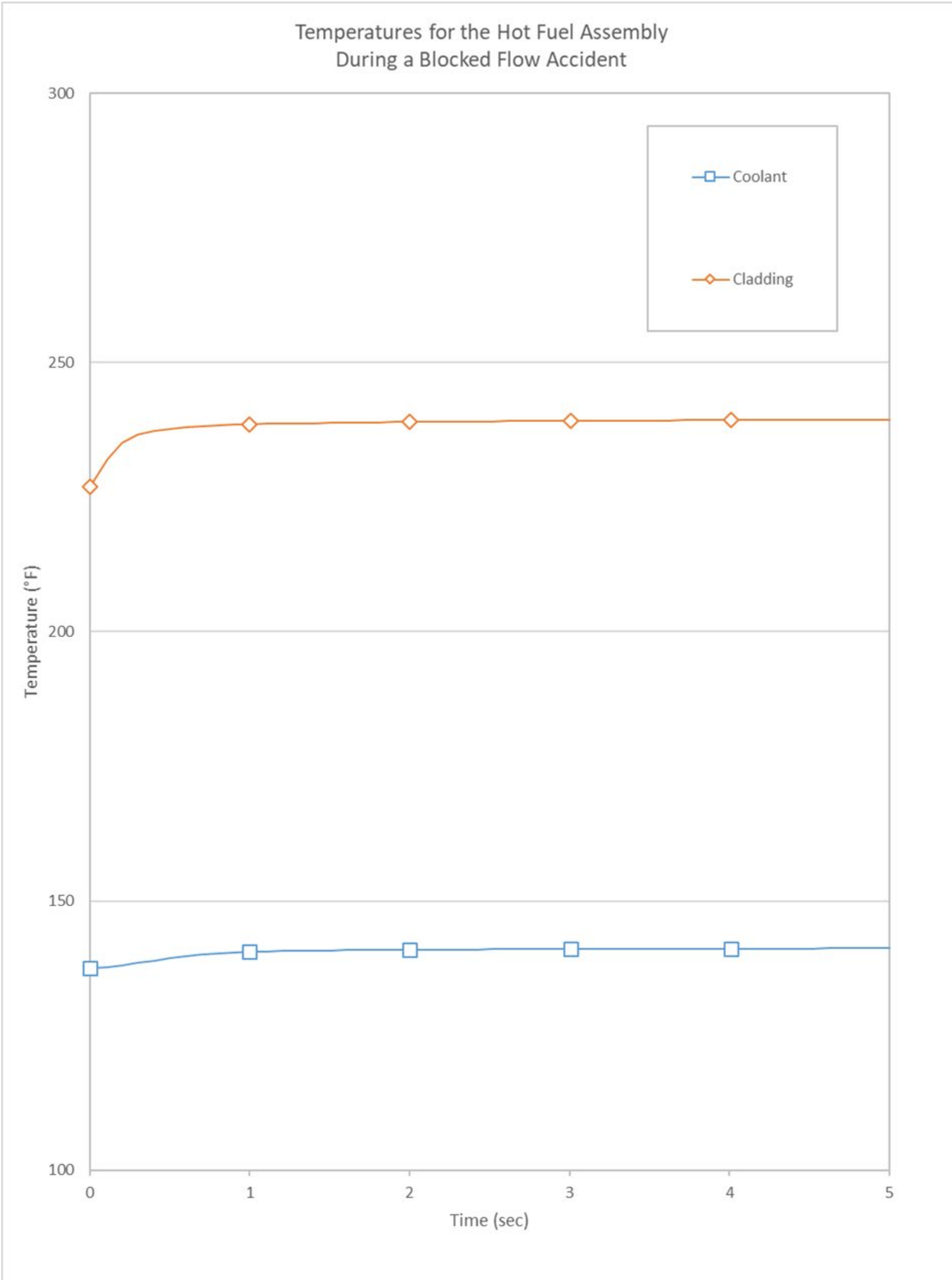


Figure 4-11 – Hot Channel Fuel Assembly Temperatures for a Blocked Flow Accident

4.3.1.2. Loss of Flow Accident

In the event that the primary flow is interrupted by failure of the primary coolant pump or other causes, a flapper valve on the plenum which is located directly under the grid plate, will open and provide a path for natural convection cooling to be established within the reactor pool. The flapper valve is held in a closed position by the differential pressure created by the coolant flowing through the core under forced convection and the pool static head at the plenum level. The flapper is normally in the closed position during forced convection flow cooling.

Even though the PULSTAR reactor is designed to change passively to natural convection flow when forced flow ceases, there is a transient period before natural convection flow can be established and start to remove decay heat. The analysis for this transient is to show that the peak cladding temperature remains at acceptable values and that the maximum credible loss-of flow accident would not result in a degradation of fuel integrity.

The initiating event for this transient scenario is stoppage of the primary coolant pump due to loss of electrical power while the reactor has been operating at 2.0 MWt for an extended period. The fission product activities are at saturated equilibrium, therefore decay heat following the scram will be at maximum. When the pump stops, a conservative assumption is that the pump stops instantaneously with no pump coast down. When flow reaches the setpoint of 900 gpm, and allowing for a 2.0 second flow sensor delay and a 0.05 second scram circuitry delay, the low primary flow scram then shuts down the reactor. Two scenarios are analyzed, the first where the flapper valve opens as designed, allowing natural circulation flow between the pool and reactor core to occur through the flapper valve, and a second where the flapper valve remains closed for the duration of the accident.

The simulated initial conditions for these scenarios are at the limiting safety system settings and given in Table 4-12.

Table 4-12 – Initialization – Loss of Flow Accident

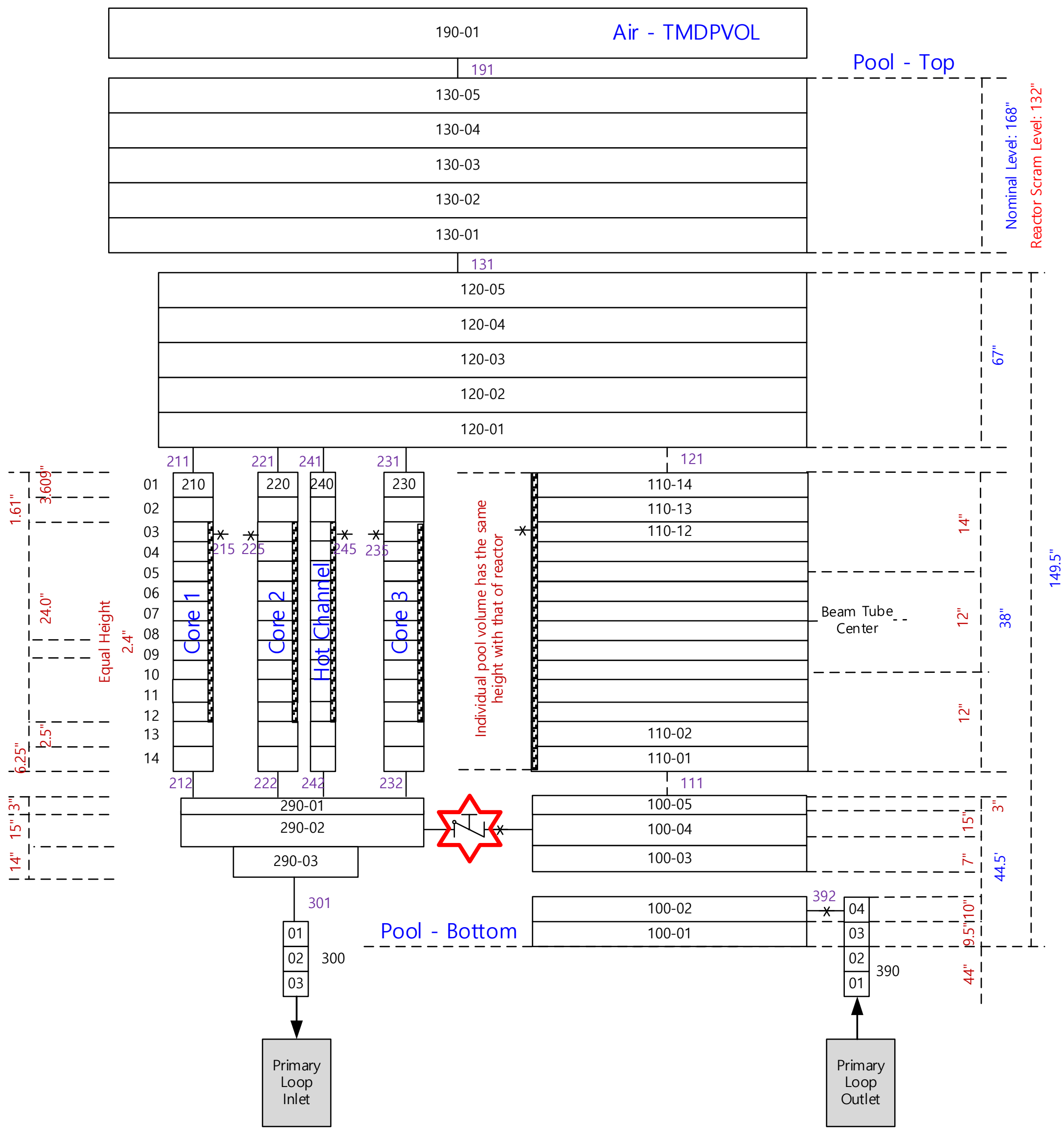
Parameter	**Nominal**	**Simulated Limiting Condition**
Reactor Power (MWt)	1.8	2.0
Power Peaking Factor	2.54	3.0
Primary Coolant Flow Rate (gpm)	1000	900
Pool Temperature (°F)	105	117
Core Temperature Rise (°F)	12.4	15.3

Secondary Coolant Flow Rate (gpm)	1000	1000
Secondary Coolant Temperature (°F)	92	98.82
Secondary Coolant Pressure (psig)	18.5	18.5
Pool Level from Top of Core (inches)	240	204
Peak Fuel Centerline Temperature (°F)		1110.8
Peaking Cladding Temperature (°F)		226.9

4.3.1.2.1. RELAP Models and Assumptions for Loss of Flow Accident

RELAP5 nodalization of the PULSTAR Reactor and Coolant System is given in Figure 4-12 and Figure 4-13 and key modeling features are summarized below and transient analysis is carried out until stable core cooling is achieved:

- The transient is initiated by the complete stop of the primary coolant pump from full power limiting conditions as given in Table 4-12.
- Reactor scram occurs with transient initiation with a total time delay of 2.05 sec.
- No coast down of the pump is assumed.
- Flapper valve is either modeled to open by the loss of differential pressure across the valve, that is, differential pressure between the reactor plenum and the adjacent pool, or to remain closed depending on the scenario.
- Secondary cooling of the primary loop heat exchanger is assumed to be available.
- Reactor Core and Hydrodynamics Model
    - The core consists of 4 fuel channels submerged in the pool. An additional hot fuel pin is imbedded in the hot fuel channel. See Figure 4-12.
    - Flow communication occurs between the fuel channels and the pool through the fuel channel top opening and four bypass flow holes at upper part of a fuel channel.
    - Convection heat transfer is modeled from the fuel to the fuel channel to the fuel box then to the pool.
- Core Power Generation Model
    - Core power distribution provided by NCSU with conservative peaking factor of 3.0. [2]
    - Core decay heat is modeled using ANSI/ANS 5.1-1979 multiplied by 1.0. [7]



Core Channels

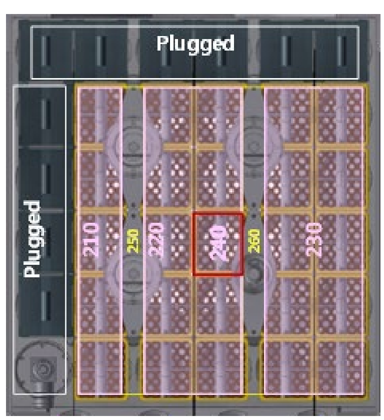


Figure 4-12 – RELAP5 Reactor and Pool Nodalization for the Loss of Flow Accident

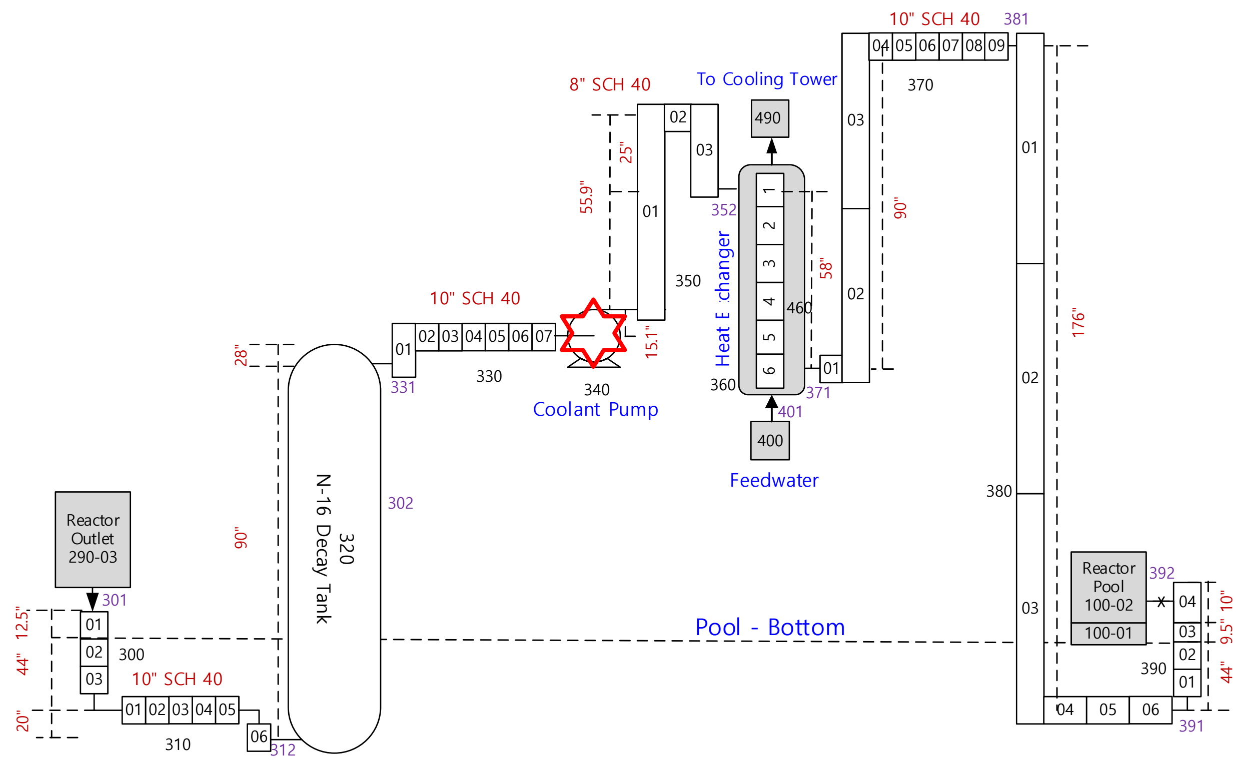


Figure 4-13 – RELAP5 Coolant System Nodalization for the Loss of Flow Accident

4.3.1.2.2. Analysis Results for Loss of Flow Accident

During the first two seconds of the transient, as shown in Figure 4-14, the reactor power decreases as the result of negative reactivity feedback caused by reduction in moderator density as the coolant heats up before the reactor scram. The reactor scrams when the setpoint for low coolant flow rate is reached (900 gpm).

Figure 4-15 shows the transient behavior of the primary coolant flow rate in fuel channels for the flapper open scenario, while Figure 4-16 shows the transient behavior of the counter current flow between fuel channels for the flapper closed scenario. The power/cooling mismatch causes some local boiling and flow oscillations during the transition to stable natural convection cooling. After transition, stable upward flow in all fuel channels is established for the flapper open scenario, while counter current flow between fuel channels is established for the flapper closed scenario. Negative flow rates signify upward flow.

Figure 4-17 shows the transient behavior of the natural circulation flow rates for the flapper open scenario. Natural circulation is fully established through the flapper valve and primary loop within 30 seconds from the start of the transient and ensures long term stable cooling of the core with no excessive heat up of the fuel. Flow through the flapper valve from the reactor pool splits at the reactor outlet plenum, that is, upward flow to the core and downward flow to the primary coolant loop that returns to the reactor pool.

Figure 4-18 shows the transient behavior of the natural circulation flow rates for the flapper closed scenario. Counter-current flow between the fuel channels is fully established through the core within 30 seconds from the start of the transient as shown in Figure 4-16, even if there is no flow through the flapper valve. Downward flow from the core to the primary coolant loop is negligible.

Figure 4-19 and Figure 4-20 shows the transient behavior of the hot spot fuel and cladding temperatures. Fuel pin temperature does not exceed the maximum fuel pin temperature of 1110.8 °F seen during steady-state operation at limiting conditions for both the flapper open and flapper closed scenarios (see Section 4.1). Maximum cladding temperature does increase to 323.6 °F and 329.6 °F for the flapper open and closed conditions, respectively, however, all temperatures are well below established safety limits.

The sequence of events for the loss of flow accident is summarized in Table 4-13 below.

Table 4-13 – Sequence of Events – Loss of Flow Accident

Event	Flapper Valve Open (sec)	Flapper Valve Closed (sec)	Remarks
Primary Pump Stops	0.0	0.0	Initiating event
Maximum Hot Pin Temperature	0.0 (1110.8 °F)	0.0 (1110.8 °F)	
Flapper Valve Opens	1.5	NA	Due to loss of differential pressure
SCRAM Signal	2.0	2.0	Low flow signal + 2.0 second delay
Control Rods Start to Drop	2.05	2.05	0.05 second SCRAM circuit time
Rods Reach Bottom	3.05	3.05	1.0 second rod drop time
Onset of Core Flow Reversal	3.5	5.0	
Hot Channel Boiling	4.5	5.5	
Maximum Cladding Temperature	4.5 (323.6 °F)	5.5 (329.6 °F)	
Time to Stable Natural Circulation	~100	NA	

4.3.1.2.3. Summary of Loss of Flow Accident

A loss of flow while operating at 2.0 MWt poses no hazards to the safety of the reactor, even in the event that the flapper fails to open. The maximum resulting temperatures are well below the safety limits.[a][3] See Table 4-14.

Table 4-14 – Summary – Loss of Flow Accidents

Parameter	**Flapper Position**		**Remarks**
	Open	**Closed**	
Peak Fuel Temperature (°F)	1110.8	1110.8	Safety Limit – 4352 °F
Peak Cladding Temperature (°F)	323.6	329.6	Safety Limit – 2200 °F

[a]Maximum cladding temperature in NUREG 1537 Part1 Appendix 14.1 is listed as 1500 °C (2732 °F) for PULSTAR fuel. The cladding temperature limit of 1200 °C (2200 °F) along with a 17% cladding oxidation limit was set based on discussion with the NRC via teleconference followed up by an email on August 1, 2018.

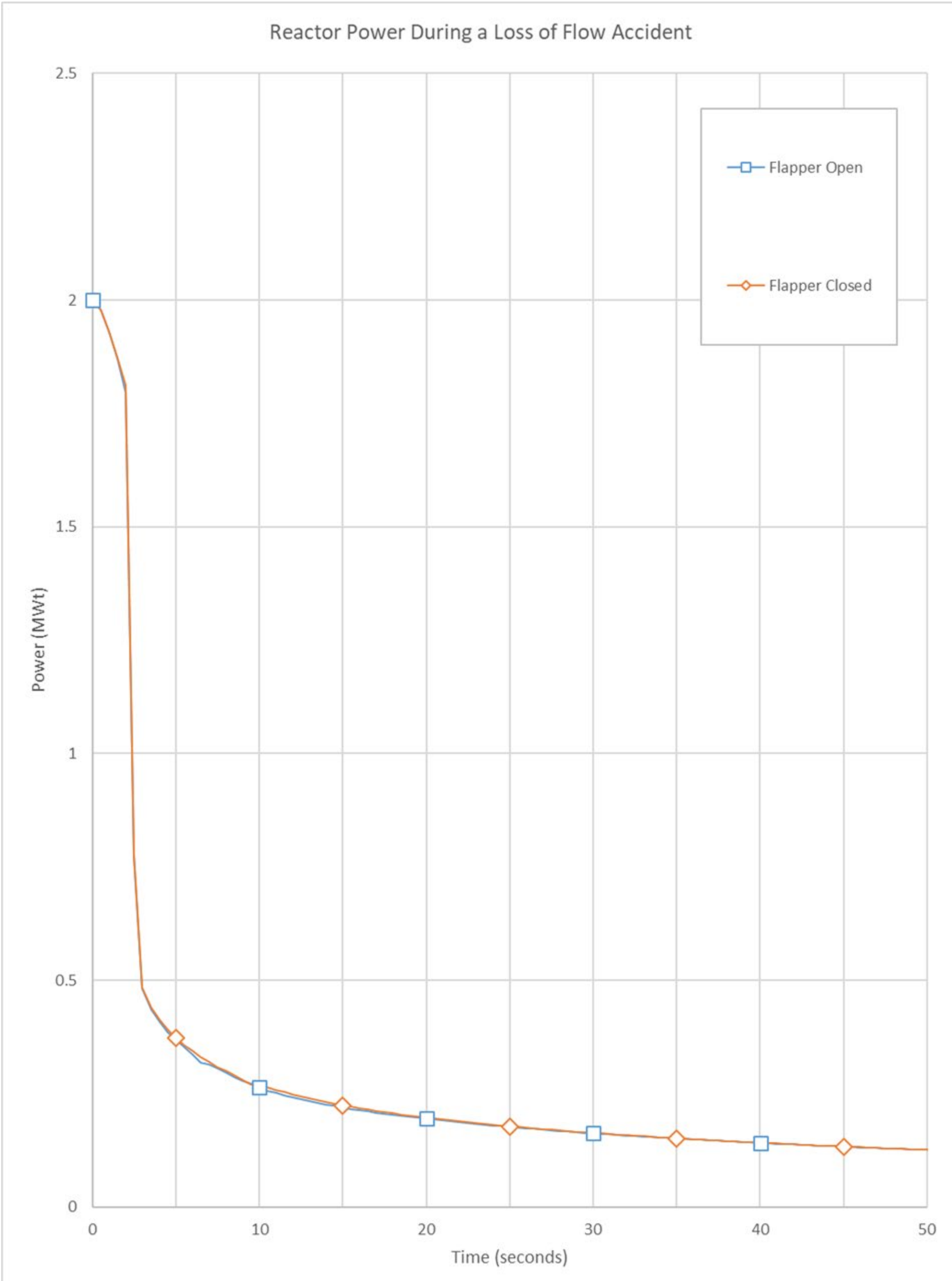


Figure 4-14 – Reactor Power – Loss of Flow Accident

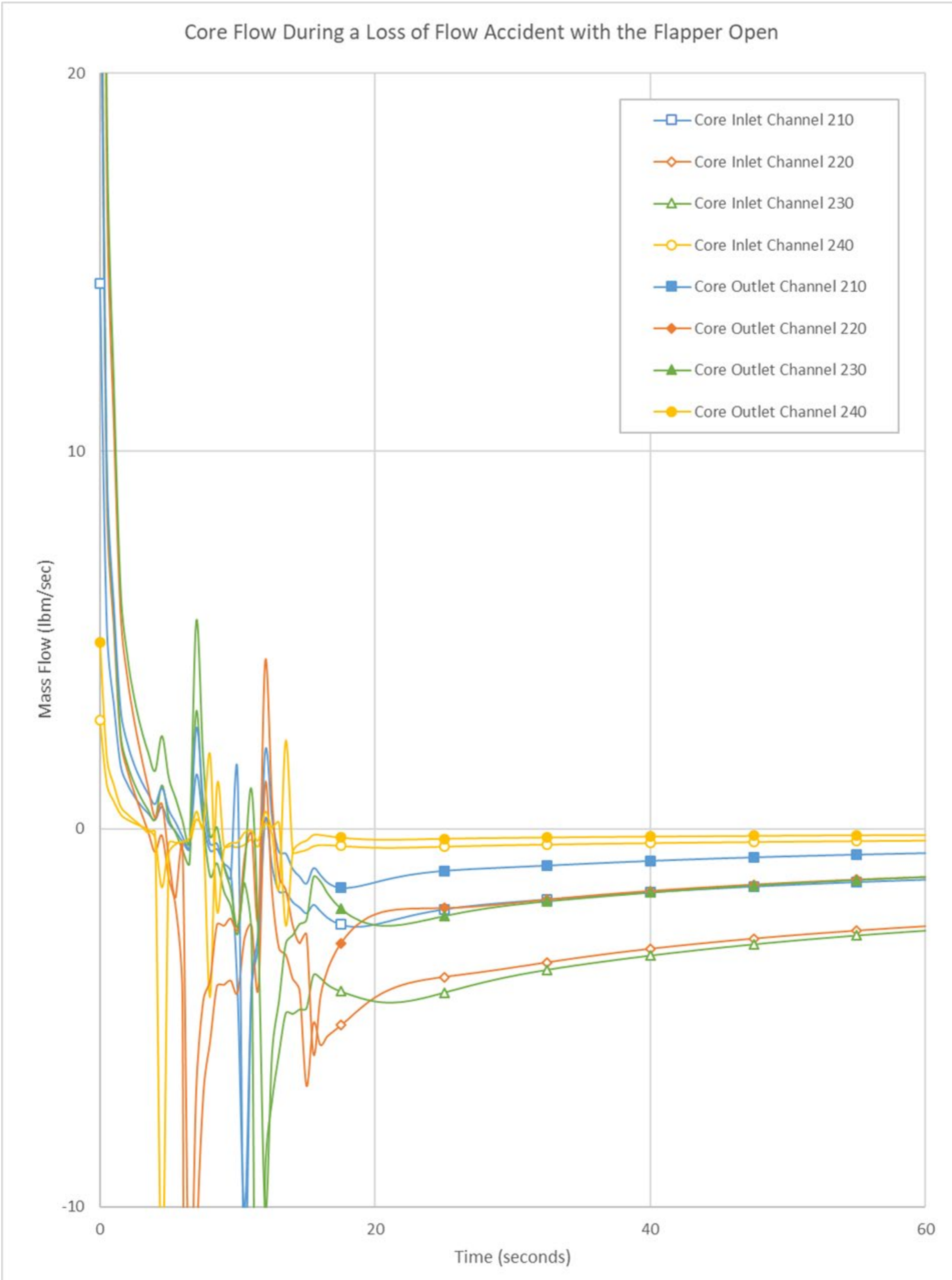


Figure 4-15 – Fuel Channel Coolant Flow with Flapper Open – Loss of Flow Accident

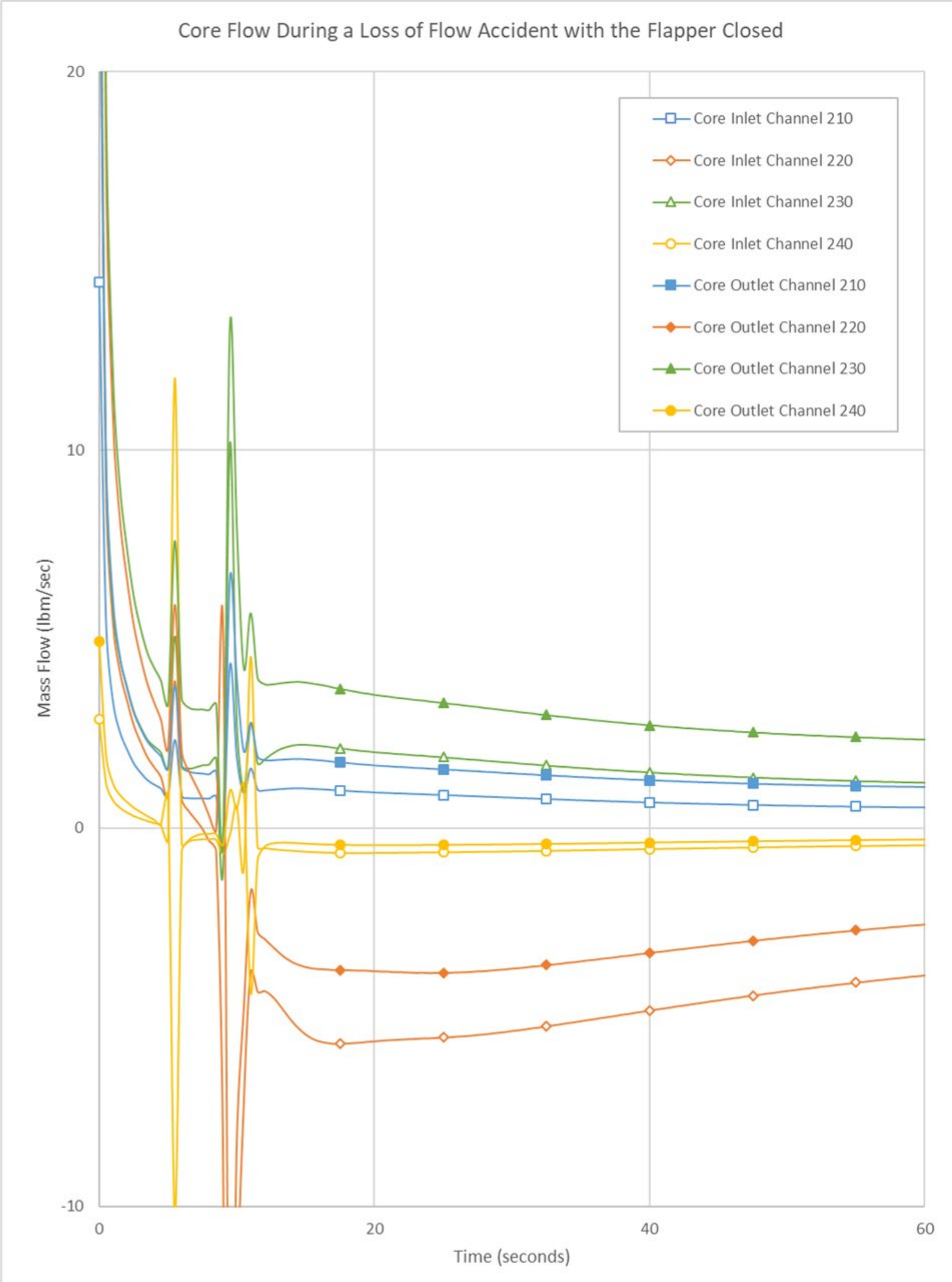


Figure 4-16 – Fuel Channel Coolant Flow with Flapper Closed – Loss of Flow Accident

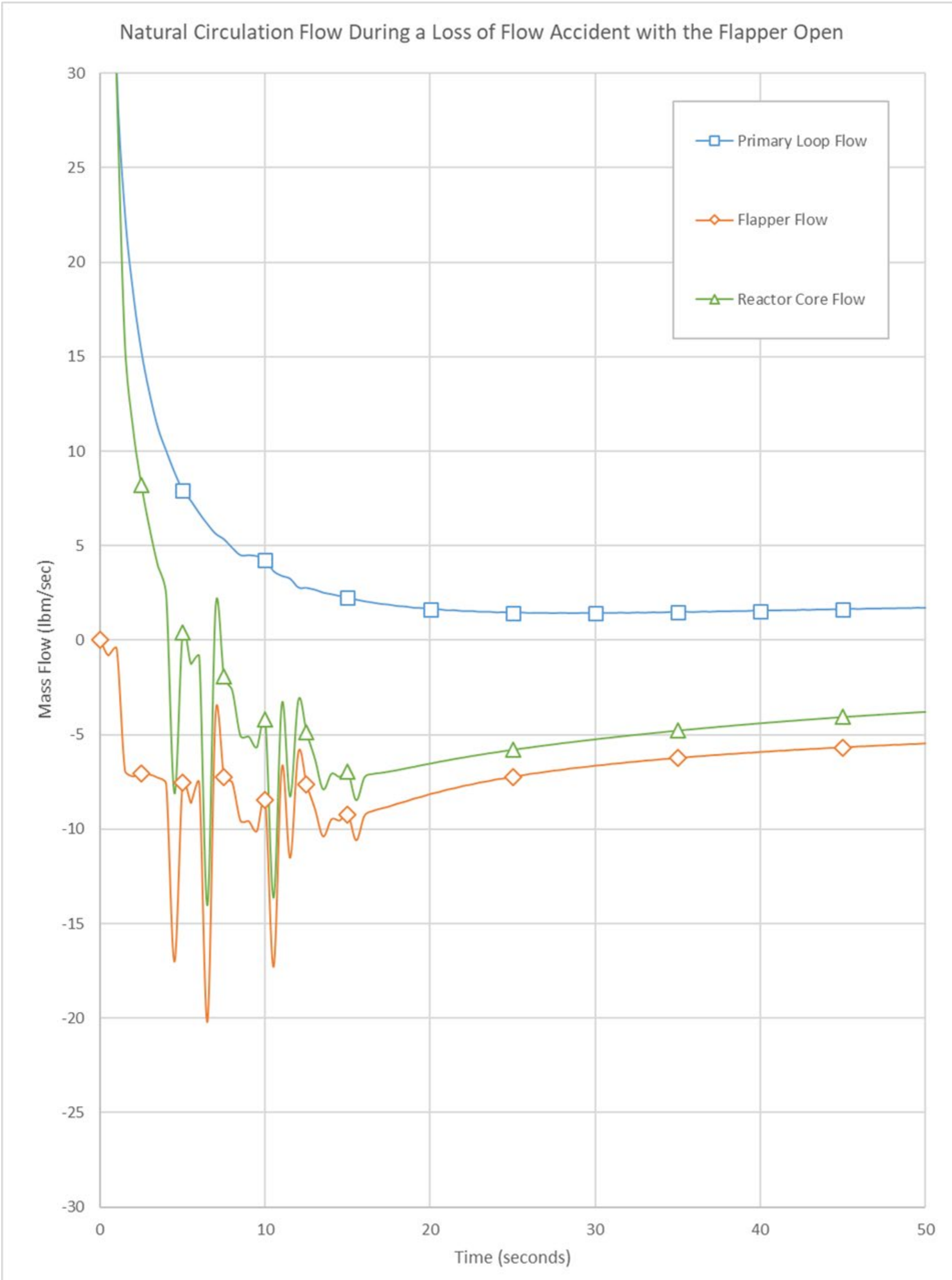


Figure 4-17 – Natural Circulation Flow Rate with Flapper Open – Loss of Flow Accident

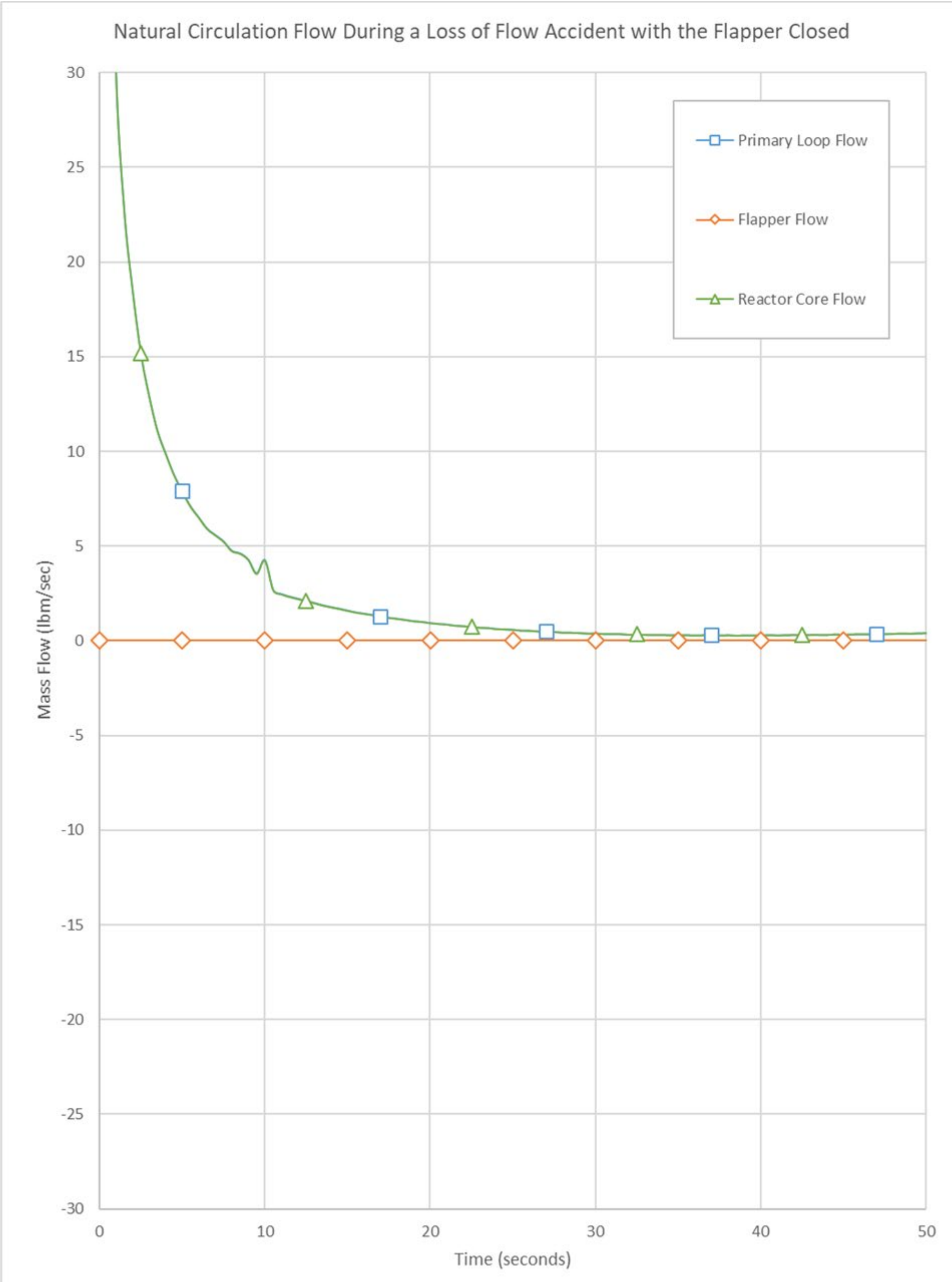


Figure 4-18 – Natural Circulation Flow Rate with Flapper Closed – Loss of Flow Accident

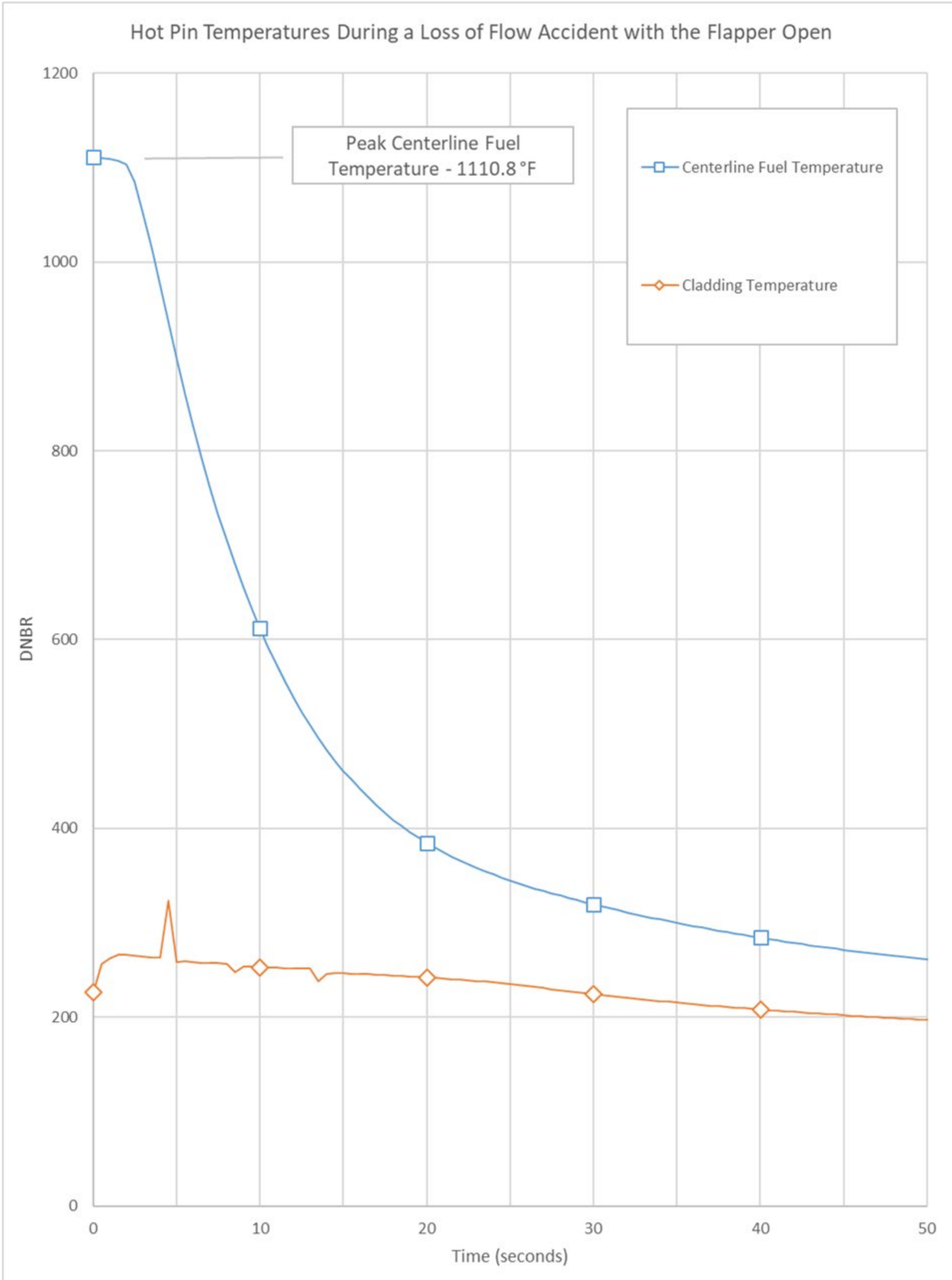


Figure 4-19 – Hot Pin Temperatures with Flapper Open – Loss of Flow Accident

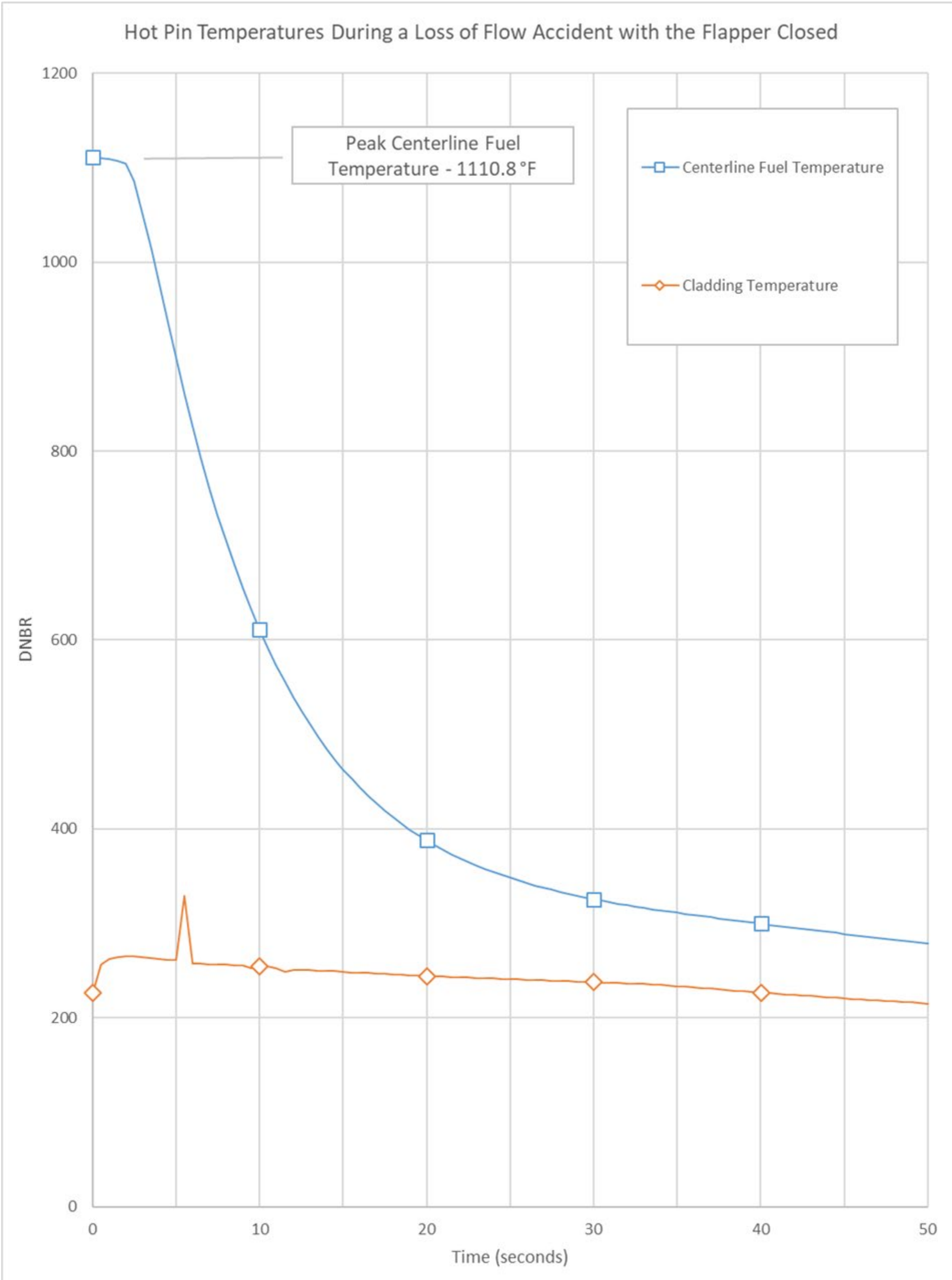


Figure 4-20 – Hot Pin Temperatures with Flapper Closed – Loss of Flow Accident

4.3.2. Loss of Coolant Accident

In many non-power reactor designs, the loss of coolant accident (LOCA) is of no consequence because decay heat in the fuel is so small as to be incapable of causing fuel failure. In some higher power reactors an engineered safety feature, such as an emergency core cooling system, may need to be operable for some time after reactor shutdown to remove decay heat in the event of a LOCA. At the PULSTAR reactor, possible initiators of LOCA events are the following:

- failure or malfunction of a component in the primary coolant loop
- failure or malfunction of an experimental facility, such as a beamtube
- failure of the reactor coolant boundary

A loss of coolant accident resulting in the complete or partial uncovering of the reactor core is possible since the pool liner has several large penetrations, and reactor coolant pipes are located at elevations that would permit compete drain down in the event of a failure. The postulated breaks are large enough that a loss of coolant could occur in a short period of time while a significant amount of heat is still being generated in the fuel pins from fission product decay heat.

While investigating the consequences of such an accident, it was assumed that the reactor had been operating with a 25 fuel assembly core at a power level of 2.0 MWt for an infinite time so that fission product activities have attained saturated equilibrium. The objective is to assess impacts on reactor safety following loss of coolant while the reactor is initially operating at full power. As shown in Figure 4-21, assessments were performed for three hypothetical break locations and sizes:

(1) Pool inlet: 10-inch diameter break in the valve pit.

(2) Reactor outlet: 10-inch diameter break in the valve pit.

(3) Beamtube No.6: 12-inch × 12-inch square break to the reactor bay.

All break scenarios assume catastrophic failure. Pipe break scenarios are double-ended guillotine breaks that cannot be isolated. Beamtube No.6 break scenario is complete guillotine break on the pool side with a completely unobstructed opening on the reactor bay side and is analyzed both with the flapper open and with the flapper closed.

The analysis determines if the peak fuel cladding temperature will remain below the safety limit of 2200 °F[a], which will ensure the integrity of the fuel cladding.

4.3.2.1. RELAP Models and Assumptions for Loss of Coolant Accident

RELAP5 nodalization of the PULSTAR system is given in Figure 4-21 and key modeling features are summarized below and transient analysis is carried out until stable core cooling is achieved after turn-around of peak fuel temperature:

- Three break locations are modeled and each scenario is initiated from the simulated limiting conditions as given in Table 4-15.
- Break Boundary Models

[a]Maximum cladding temperature in NUREG 1537 Part1 Appendix 14.1 is listed as 1500 °C (2732 °F) for PULSTAR fuel. The cladding temperature limit of 1200 °C (2200 °F) along with a 17% cladding oxidation limit was set based on discussion with the NRC via teleconference followed up by an email on August 1, 2018.

    - Pool Inlet break discharges primary coolant to a downstream pit through a 10-inch diameter double-ended break.
    - Reactor Outlet break discharges primary coolant to a downstream pit through a 10-inch diameter double-ended break.
    - Beamtube No.6 break discharges primary coolant to the reactor bay floor through 12x12 inch square break.
- Primary pump stops at the reactor scram signal.
- Flapper valve is either modeled to open by the loss of differential pressure across the valve, that is, differential pressure between the reactor plenum and the adjacent pool or, for Beamtube No.6 break, to remain closed depending on the scenario.
- Metal water reactions is turned off for the pool inlet and reactor outlet break scenarios since the environment is almost completely air filled.
- Metal water reactions is turned on for the Beamtube No.6 break scenario since the environment is almost completely steam filled.
- Reactor Core and Hydrodynamics Model
    - The core consists of 4 fuel channels submerged in the pool. An additional hot fuel pin is imbedded in the hot fuel channel.
    - Convection heat transfer is modeled at the heat structure surfaces of the fuel rods and fuel boxes.
    - 2-D radial and axial heat conduction is modeled in the fuel and fuel box using the reflood model.
    - Radiation heat transfer is modeled from the fuel to the fuel box then to the pool wall.
- Core Power Generation Model
    - Core power distribution provided by PULSTAR MCNP model with conservative peaking factor of 3.0. [2]
    - Core decay heat is modeled using ANSI/ANS 5.1-1979 multiplied by 1.0. [7]

Table 4-15 – Loss of Coolant Accident – Initialization

Parameter	**Nominal**	**Simulated Limiting Conditions**
Reactor Power (MWt)	1.8	2.0
Power Peaking Factor	2.54	3.0
Primary Coolant Flow Rate (gpm)	1000	900
Pool Temperature (°F)	105	117
Core Temperature Rise (°F)	12.4	15.3
Secondary Coolant Flow Rate (gpm)	1000	1000
Secondary Coolant Inlet Temperature (°F)	92	98.8
Secondary Pressure (psig)	18.5	18.5
Pool Level from Top of Core	240	204

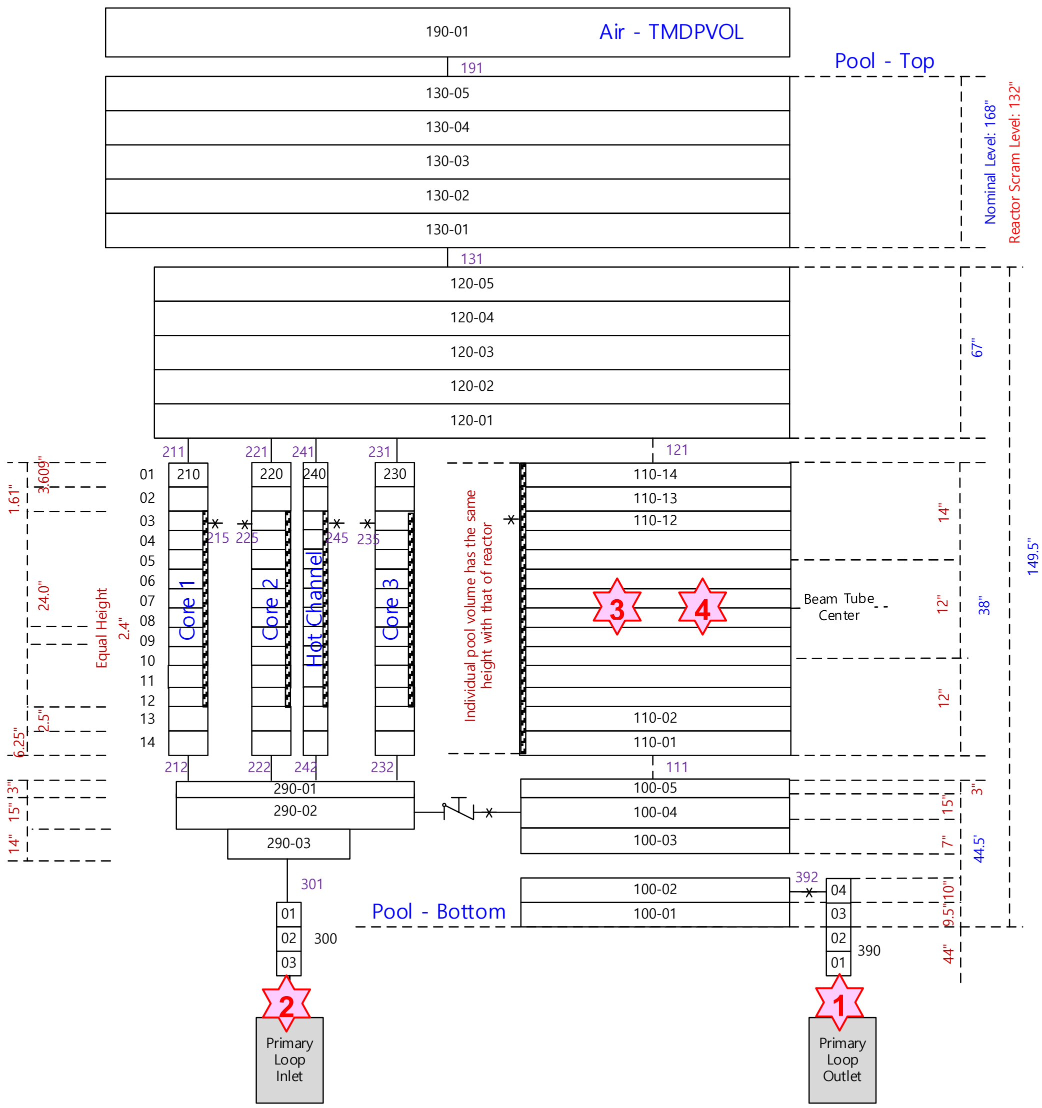


Figure 4-21 – RELAP5 Nodalization and Break Locations Modeled for Loss of Coolant Accident Events

4.3.2.2. LOCA Analysis Results

The accidents modeled to have the worst potential consequences are those initiated by the catastrophic double-ended guillotine breaks in the main 10-inch diameter reactor coolant piping located in the vault pit prior to the pool isolation valves, or a catastrophic guillotine break of Beamtube No.6, the largest beamtube at the PULSTAR. All scenarios assume that the reactor is operating at full licensed power and has been operating long enough for the fuel to contain fission products at equilibrium concentrations. Therefore, the maximum possible decay heat is available at the start of the event. All scenarios assume that the pool level is at the limiting condition. The low pool level initiates a reactor scram and shuts down the reactor for all three scenarios. Table 4-16 summarizes the time sequence of events and Table 4-17 lists the initial break flows and times to uncover the core.

Table 4-16 – Sequence of Events – Loss of Coolant Accident

Event	**Break Location**				**Remarks**
	Pool Inlet (sec)	**Reactor Outlet (sec)**	**Beamtube No.6 (sec)**		
			Flapper Open	**Flapper Closed**	
Break Initiation	0.0	0.0	0.0	0.0	Initiating Event
Reactor Scram signal and Primary Pump Trip	0.0	0.0	0.0	0.0	Primary pump trips at scram setpoint
Scram Rods start to drop	2.05	2.05	2.05	2.05	Scram time delay of 2.05 second
Full Insertion of Scram Rods	3.05	3.05	3.05	3.05	1.0 second rod drop time
Flapper Valve fully Open	92.5	747.5	0.9	N/A	Opens on loss of differential pressure
Core† starts to Uncover	101.5	227.0‡	95.0	95.0	77.25 inches from pool bottom
Core† fully voided	113.5	229.0‡	N/A	N/A	53.25 inches from pool bottom
Equilibrium Pool Level	~160.0	~1200.0	~200	~200	
Peak Fuel Temperature (°F)	2985.0 sec (2031.7 °F)	3165.0 sec (1983.1 °F)	2022.5 sec (1635.6 °F)	2502.5 sec (1849.8 °F)	Safety Limit 4352 °F
Peak Cladding Temperature (°F)	2995.0 sec (2011.8 °F)	3215.0 sec (1964.7 °F)	2065.0 sec (1619.1 °F)	2500.0 sec (1832.5 °F)	Safety Limit 2200 °F

Notes:
†The core is defined as the top of the active fuel to the bottom of the active fuel.
‡For the Reactor Outlet Break, the water level in the core and pool water level are independent of each other, see Figure 4-25 and Figure 4-26.

Table 4-17 – Break Flow Parameters – Loss of Coolant Accident

Event	Break Location			
	Pool Inlet	Reactor Outlet	Beamtube No.6	
			Flapper Open	Flapper Closed
Cross-sectional Break Area (in^2)	78.5	78.5	144	144
Initial Break Flow Rate (lbm/sec)	996	505	1301	1301

Figure 4-22 shows the transient behavior of the total break flow rate. The initial flow rate is larger for the beamtube break because of the larger cross-sectional break area. Break flow terminates when the break upstream becomes fully voided or when the pressure across the break equilibrates.

Figure 4-23 shows the transient behavior of reactor power. A low pool water level scram shuts down the reactor when the LSSS setpoint of 204 inches of water above the core is reached. Reactor power varies by reactivity feedback from moderator temperature and density changes before reactor scram. Reactor power increases in the reactor outlet break where core gets overcooled driven by break flow, while it decreases in other breaks. Figure 4-24 shows the corresponding transient behavior of total core reactivity. Reactivity oscillation in Beamtube No.6 break is due to moderator density feedback, since the core is partially filled with water and the core level oscillates.

Figure 4-25 shows the transient behavior of the pool water levels. For the breaks occurring at the reactor outlet and pool inlet piping, the water inventory in the core and the pool depletes to eventually completely uncover the core. For the reactor outlet break, because the fuel assemblies are in boxes, once the water drops below the top of the fuel assembly the water level in the core becomes independent of the pool water level and depletes quickly (see Figure 4-26). Therefore, the core starts to uncover at 227 seconds and is completely uncovered at 229 seconds, while the pool water level at the same elevation depletes at a much slower rate. For all other break scenarios, the core water level and pool water level remain coupled at all times.

The rate of uncovering the core is slower for the reactor outlet break due to a smaller break flowrate, and inflow from the pool to the core before core top is uncovered. If the break occurs in Beamtube No.6, pool and core water levels are maintained at the level of break elevation so that the bottom 4 inches of the fuel remains covered.

Figure 4-27 shows the air quality at the hot spot representing the fractional moisture content in the air. In the pool inlet and reactor outlet scenarios the pool has completely drained down leaving the hot spot environment mostly air filled, therefore the metal to water reaction is not evaluated. In the Beamtube No.6 break scenarios where the break elevation leaves the bottom 4 inches of the fuel covered, the environment is mostly steam filled and the metal to water reaction is evaluated. Table 4-18 lists the

cladding oxidation for the two Beamtube No.6 break scenarios. The resulting oxidation is well below the maximum cladding oxidation limit of 17%.[a]

Figure 4-28 and Figure 4-29 show the hot pin centerline and cladding temperatures. As shown in the figures, the fuel heats up when the core uncovers, however, temperatures start to decrease as core heat removal by the convective and radiation heat transfer exceeds core decay heat. Maximum temperatures remain below the safety limits.

Table 4-18 – Cladding Oxidation – Loss of Coolant Accident

Parameter	Beamtube No.6 with Flapper Open	Beamtube No. 6 with Flapper Closed
Maximum Cladding Oxidation Depth (inches)	0.0015	0.0021
Maximum Cladding Oxidation	7.4%	10.4%

4.3.2.3. Summary of Loss of Coolant Accidents

The loss of coolant accident was analyzed from three different break locations. At all times for all three accident scenarios, maximum fuel and cladding temperatures remain below the safety limits. Table 4-19 summarizes maximum temperatures for the three scenarios.

Table 4-19 – Summary – Loss of Coolant Accident

Event	Break Location				Remarks
	Pool Inlet	Reactor Outlet	Beamtube No.6		
			Flapper Open	Flapper Closed	
Peak Fuel Temperature (°F)	2031.7	1983.1	1635.6	1849.8	Safety Limit – 4352 °F
Peak Cladding Temperature (°F)	2011.8	1964.7	1619.1	1832.5	Safety Limit – 2200 °F

[a]Maximum cladding temperature in NUREG 1537 Part1 Appendix 14.1 is listed as 1500 °C (2732 °F) for PULSTAR fuel. The cladding temperature limit of 1200 °C (2200 °F) along with a 17% cladding oxidation limit was set based on discussion with the NRC via teleconference followed up by an email on August 1, 2018.

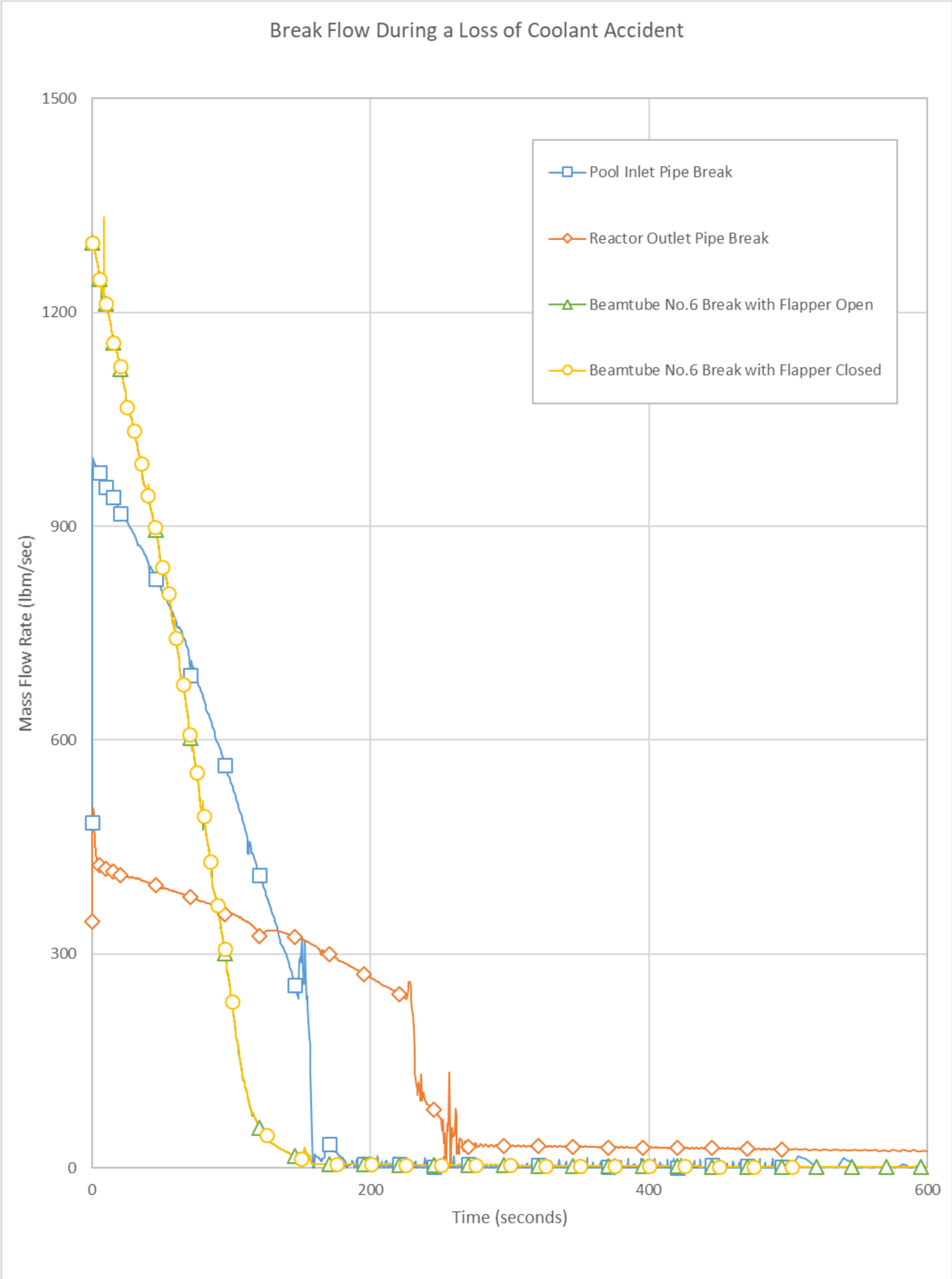


Figure 4-22 – Total Break Flow Rate – Loss of Coolant Accident

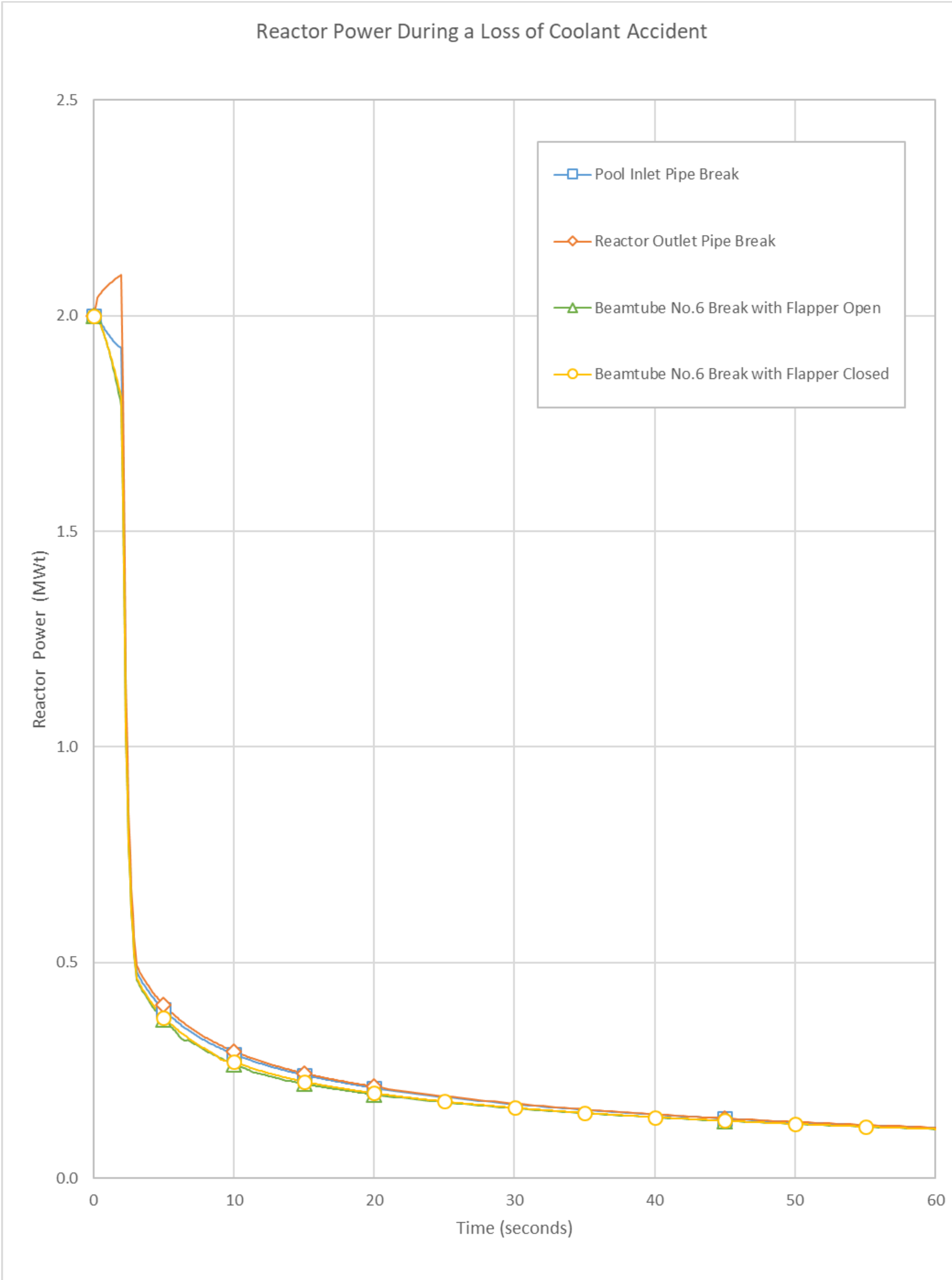


Figure 4-23 – Reactor Power – Loss of Coolant Accident

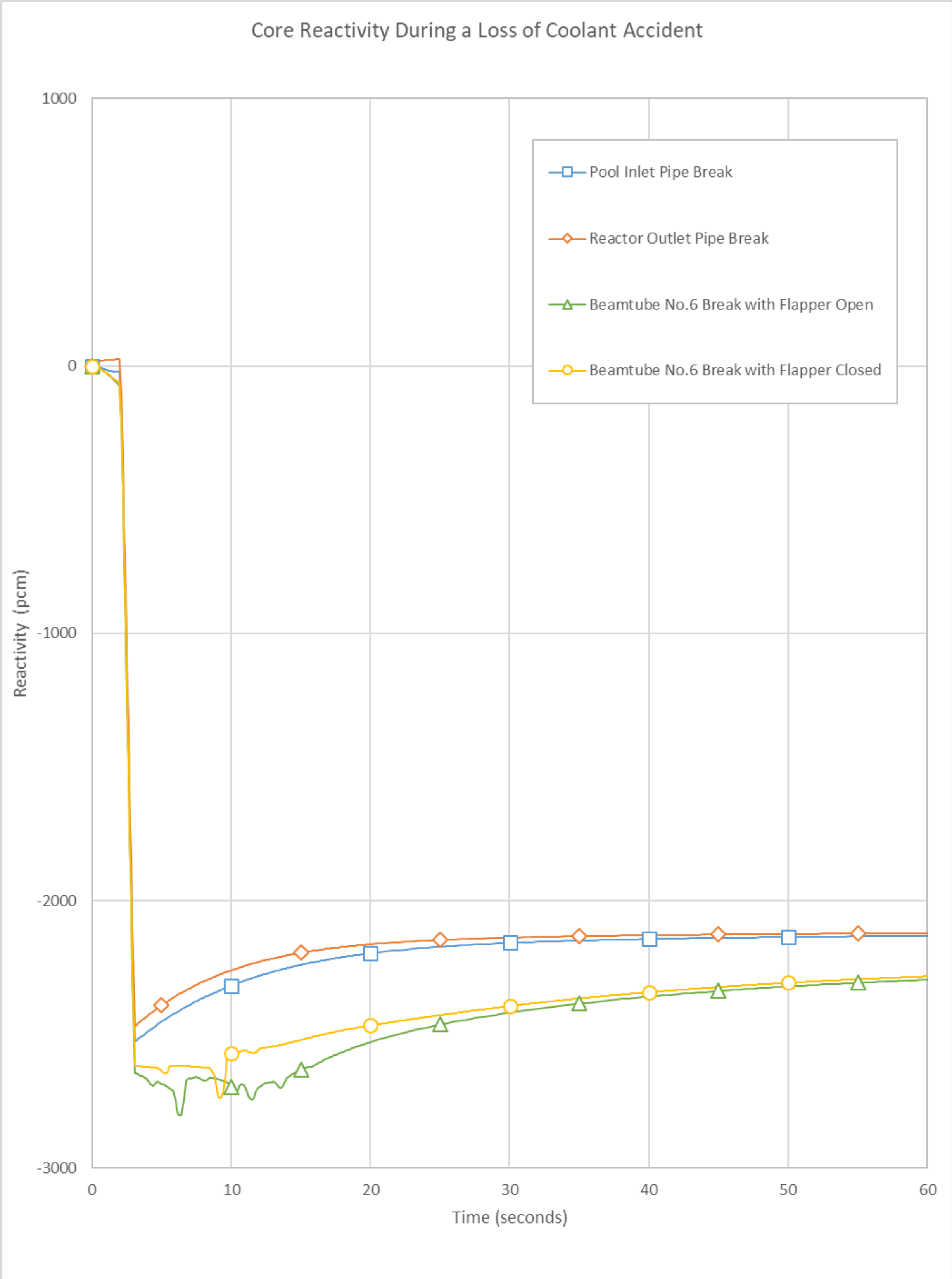


Figure 4-24 – Total Core Reactivity – Loss of Coolant Accident

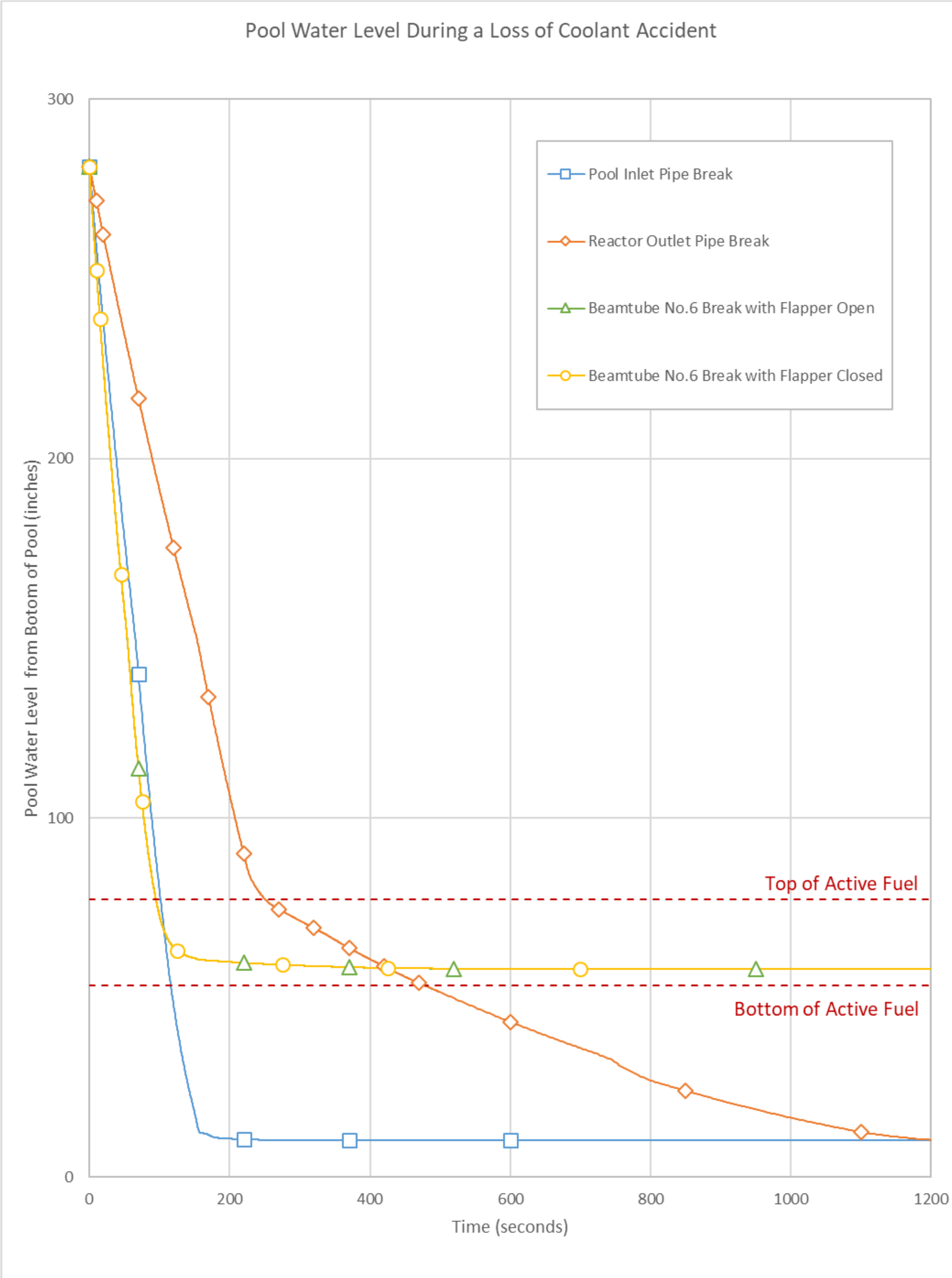


Figure 4-25 – Pool Water Level – Loss of Coolant Accident

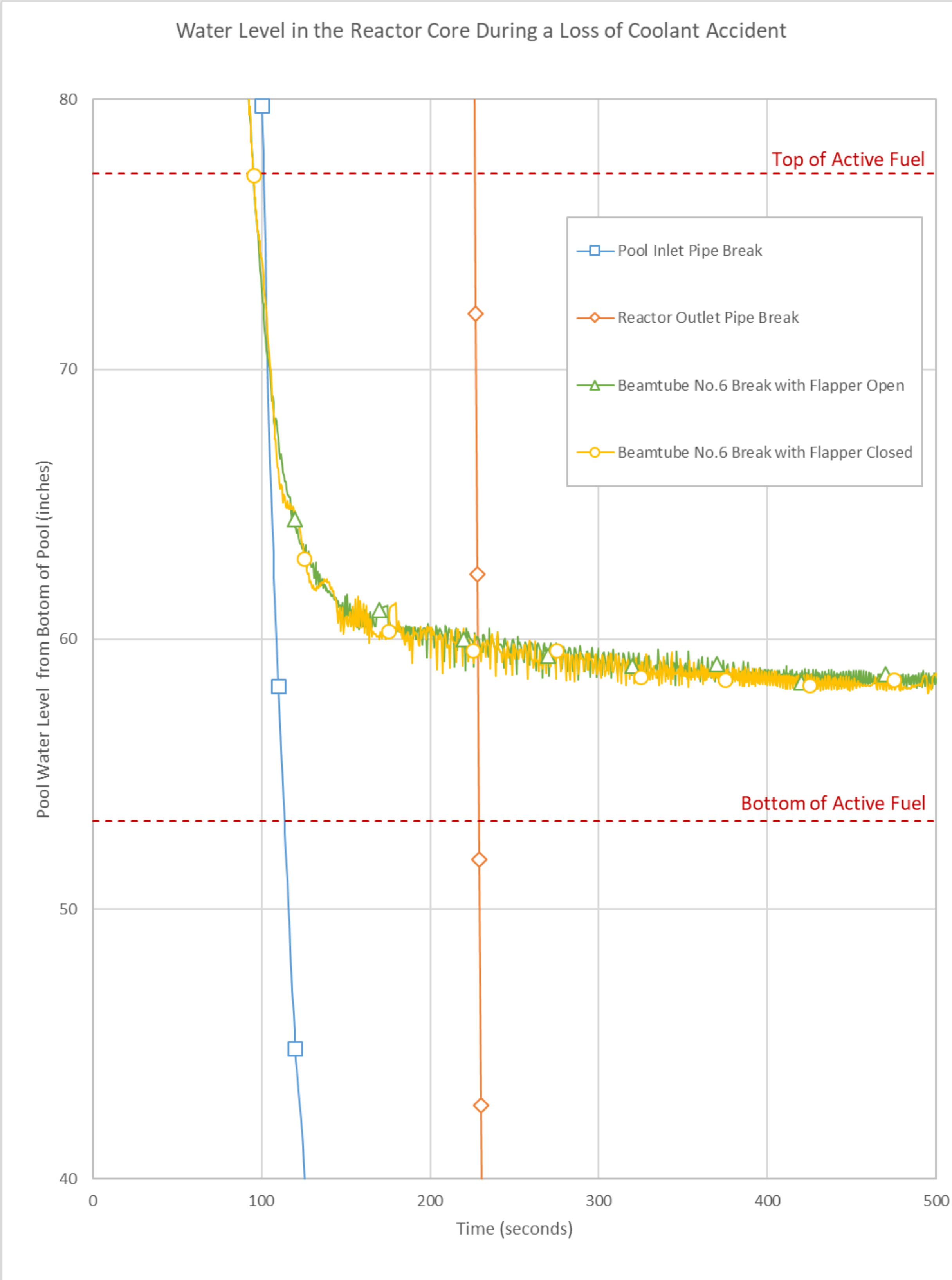


Figure 4-26 – Core Water Level – Loss of Coolant Accident

Figure 4-27 – Air Quality at the Hot Spot – Loss of Coolant Accident

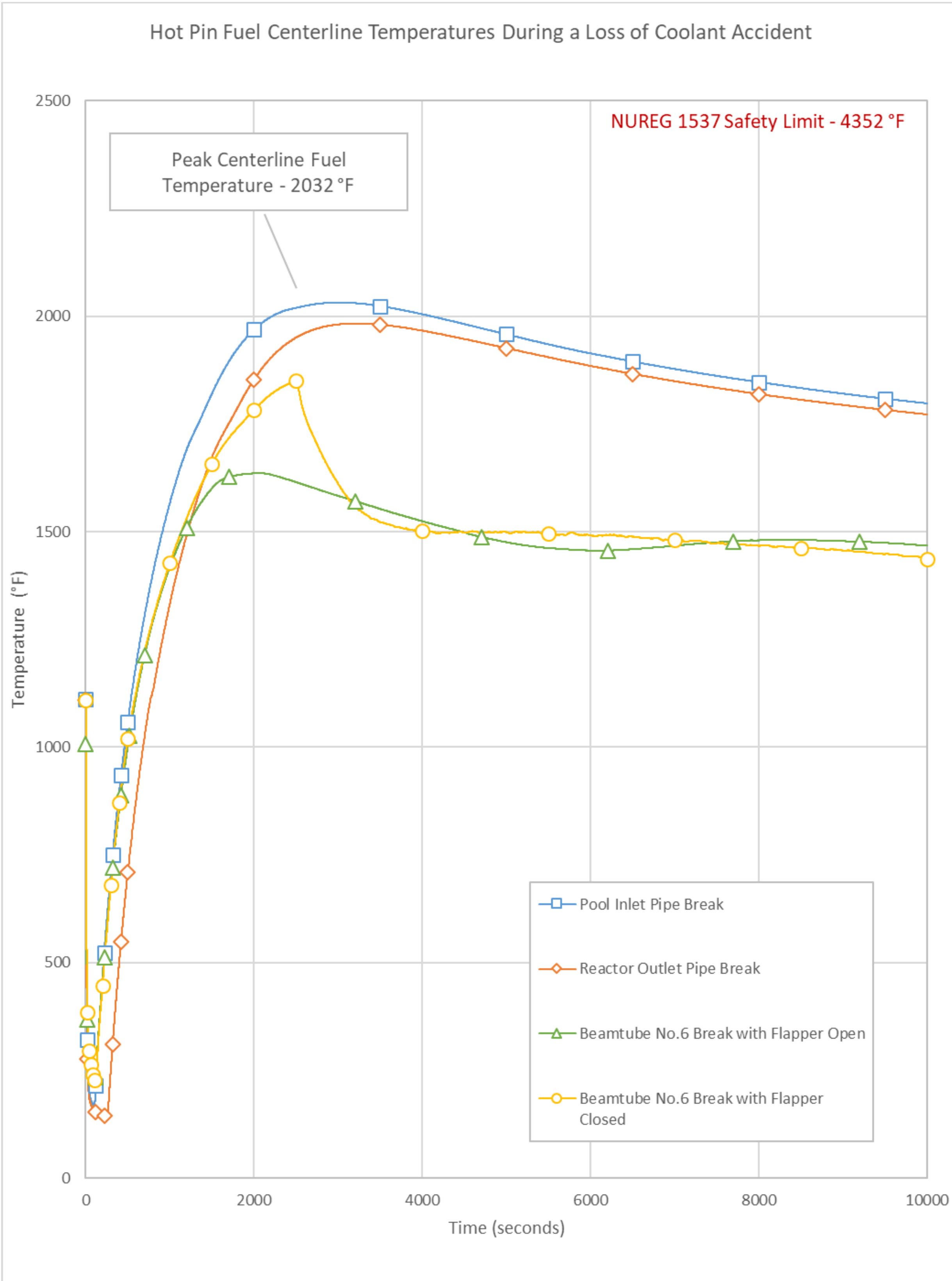


Figure 4-28 – Hot Pin Fuel Centerline Temperature – Loss of Coolant Accident

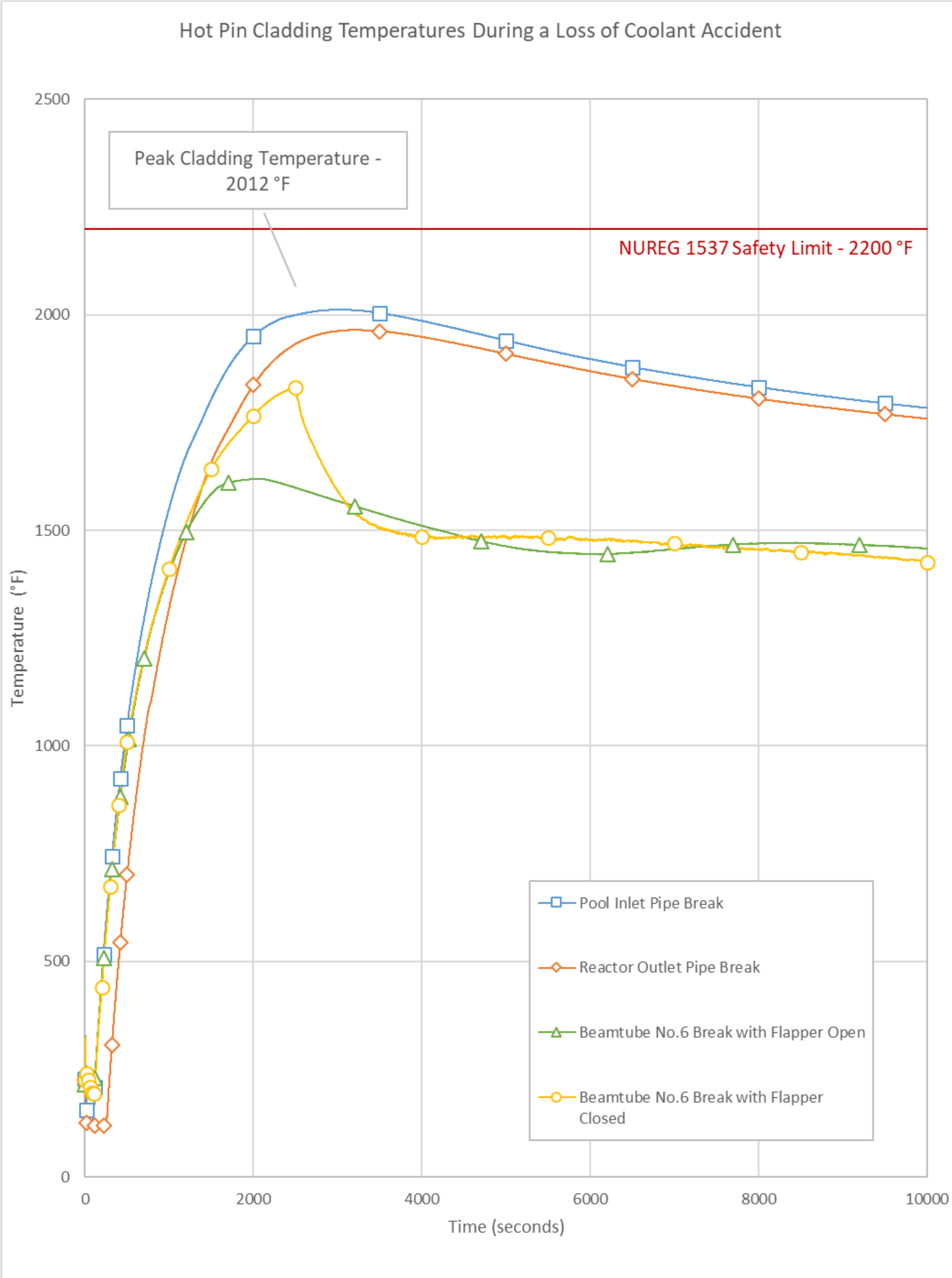


Figure 4-29 – Hot Pin Cladding Temperature – Loss of Coolant Accident

4.3.3. Reactivity Insertion Accidents

Reactivity insertion accidents can either be classified as ramp insertions or step insertions.

Ramp insertions are events where reactivity is relatively slowly but continuously inserted into the reactor. Some examples of possible initiating events for ramp insertion accidents are:

- Slow insertion of a fuel assembly or fueled experiment into a core vacancy.
- Malfunction of a control rod system.
- Operator error, especially at reactor startup.
- Malfunction of a power level indicator.
- Protracted malfunction, movement, or leakage of an experiment or experimental facility.
- Malfunction of reflector components.

A malfunction of the control rod drive mechanism or operator error during reactor startup could cause an inadvertent withdrawal of the control rod and an unplanned increase in reactor power. The accident scenario conservatively assumes that the reactor has a maximum load of fuel (consistent with the Technical Specifications) and the control system malfunction withdraws the control rods at their maximum differential reactivity worth at the maximum drive speed. The continuous removal of the rod causes a continuous decrease of reactor period and a continuous increase in reactor power. The over-power level scrams are assumed to be operable. The analysis (as discussed in Section 4.3.3.2 below) shows that the maximum power level reached during the ramp transient does not raise maximum fuel and cladding temperatures above the safety limit temperatures for fuel centerline or fuel cladding, therefore fuel integrity would be maintained.

Step insertions are events that rapidly insert large amounts of reactivity into the reactor causing a high power excursion. Some examples of possible initiating events for step insertion accidents are:

- Rapid dropping of a fuel assembly or a fueled experiment into a core vacancy.
- Rapid removal or ejection of a control rod.
- Sudden malfunction, movement, or failure of an experiment or experimental facility.
- A cold primary coolant slug.
- Malfunction of reflector components.

The accident scenario conservatively assumes that a fuel element is inadvertently dropped into a core vacancy while the reactor is critical, rapidly inserting net positive reactivity into the core. The reactor enters a prompt critical state with a net positive reactivity of 1612 pcm ($2.2), which induces a very short stable positive reactor period. Reactor power increases so quickly that the safety rods are assumed not to move significantly during the transient even though the over-power level scrams are tripped. The analysis shows that the temperature increase and formation of voids in the moderator along with the prompt negative temperature coefficient of the fuel reduces reactivity sufficiently to retard the excursion. The control rods then insert within their required drop time of 1.0 seconds, which stabilizes the subcritical reactor. The analysis below shows (as discussed in Section 4.3.3.3) that the maximum power level reached during the step insertion does not raise maximum fuel and cladding temperatures above the safety limit temperatures for fuel centerline or fuel cladding, therefore, fuel integrity would be

maintained.

4.3.3.1. RELAP Models and Assumptions for Reactivity Insertion Events

Key modeling features are summarized below and transient analysis is carried out until the core power excursion terminates:

RELAP5 nodalization of the PULSTAR system is given in Figure 4-30 and key modeling features are summarized below and transient analysis is carried out until a stable subcritical core is achieved:

- All reactivity insertion scenarios are initiated from the simulated limiting conditions from various initial power levels as given in Table 4-20 and Table 4-22.
- Reactivity Insertions
  - Ramp reactivity insertion of both 100 pcm/sec and 200 pcm/sec.
  - Step reactivity insertion of 1612 ($2.2) in 0.2 seconds.
- Primary pump remains in its predefined initial condition for the duration of the scenario.
- Flapper valve remains in its predefined initial condition for the duration of the scenario.
- Reactor Core and Hydrodynamics Model
  - The core consists of 4 fuel channels submerged in the pool. An additional hot fuel pin is imbedded in the hot fuel channel.
  - Convection heat transfer is modeled at the heat structure surfaces of the fuel rods and fuel boxes.
- Core Power Generation Model
  - Core power distribution provided by PULSTAR MCNP model with conservative peaking factor of 3.0. [2].
  - Kinetic data generated by PULSTAR MCNP model. [2]
  - Delayed neutron fraction (β) of 0.00733. [2]
  - Prompt neutron generation time of 36 μsec. [8]
  - Core decay heat is modeled using ANSI/ANS 5.1-1979 multiplied by 1.0. [7]

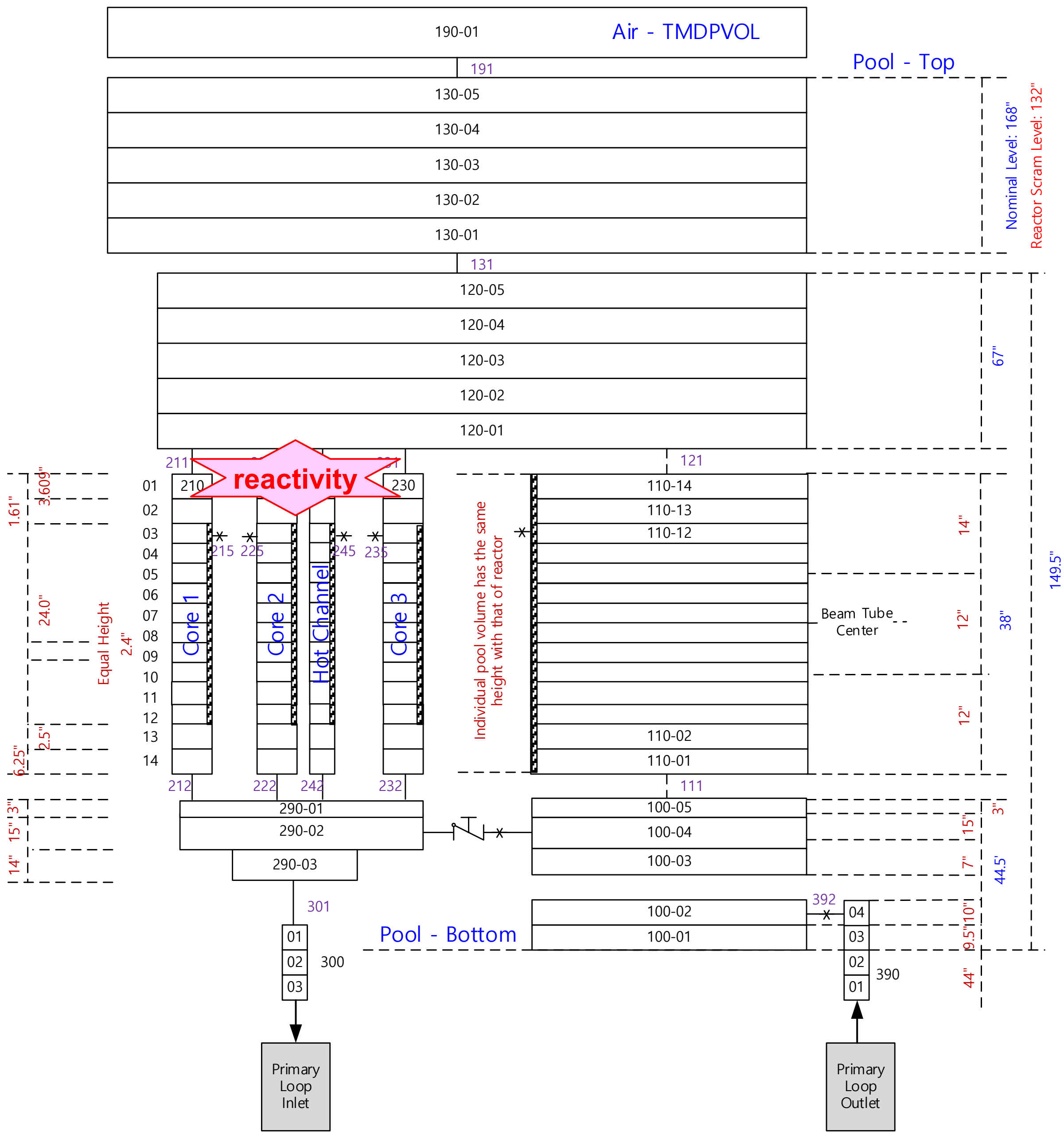


Figure 4-30 – RELAP5 Nodalization for Reactivity Insertion Accidents

4.3.3.2. Ramp Reactivity Insertion Accident

The objective is to assess impacts on reactor safety parameters, mainly fuel centerline and cladding temperatures, to ensure that fuel and cladding integrity is maintained during a ramp reactivity insertion, therefore preventing the release of fission products to the environment.

The following parameters are investigated:

- Maximum resulting power level (MWt)
- Maximum hot pin fuel centerline temperatures (°F)
- Maximum hot pin cladding temperatures (°F)

Hot zero power with forced flow is the only scenario evaluated since the hot zero power without flow would be terminated at the lower LSSS scram setpoint of 250 kWt.

Hot power refers to a primary coolant temperature at the LSSS setpoint of 117 °F, which is more limiting than the cold coolant temperature of 70 °F. For this analysis, zero power is defined to be 100 watts. All other initial conditions are at the LSSS setpoint or Technical Specification limit and are listed in Table 4-20.

It is assumed that a ramp reactivity insertion is the transient resulting from control rod withdrawal of all ganged control rods from a critical core at 100 watts through scram actuation by the over-power level scrams.

The following are assumed:

- The operator has failed to return the Linear Power Channel range selector to the most sensitive or mid-scale position. This would mean that the over-power level scrams would not be set at the most sensitive or on-scale position prior to withdrawal of the control rods, and thus an over-power level scram would occur at the scram setpoint of 2.0 MWt.
- The reactor is critical at 100 watts and operating under the conditions listed in Table 4-20.
- It is assumed that the control rods are withdrawn on gang and the gang rate at which reactivity is inserted into the core is at 100 pcm/second and 200 pcm/sec.
- The core has been loaded in the optimum configuration that places the core and associated parameters at the Technical Specification limits.
- The hot pin in the core is assumed to have a peaking factor at the Technical Specification limit of 3.0.

The event sequence is summarized in Table 4-21. The ramp insertion initiates at time equal to 0.0 seconds. The scram signal actuates at the setpoint of 2.0 MWt at time equal to 7.41 seconds and 3.93 seconds respectively for the 100 pcm/sec and 200 pcm/sec insertion scenarios. The rods begin to drop after a 0.05 second scram circuitry delay and are fully inserted at time equal to 8.46 seconds and 4.99 seconds. Control rod drop times are assumed to be at the Technical Specification limit of 1.0 seconds and ensure a stable subcritical reactor.

The analysis shows that power excursion is terminated by insertion of control rods actuated by the over-power scrams. As shown in Figure 4-31 and Figure 4-32, the negative temperature coefficient of the fuel is not effective to prevent and turnover the power increase. The excursion results in a maximum power level of 3.32 MWt and 4.96 MWt.

Figure 4-33 and Figure 4-34 shows the transient behavior of the hot pin fuel temperature and hot pin cladding temperature. The power excursion causes an initial increase in fuel temperature before the reactor scram. The slower ramp of 100 pcm/sec results in nominally higher temperatures since more energy is deposited in the fuel, resulting in a higher total integrated power. The excursion results in the maximum fuel temperatures of 161.6 °F and 139.5 °F and maximum cladding temperatures of 124.1 °F and 120.7 °F, for the 100 pcm/sec and 200 pcm/sec ramps, which are well below the safety limit of 2200 °F and 4352 °F for PULSTAR fuel.[a][3]

Table 4-20 – Initialization – Ramp Reactivity Insertion Accident

Parameters	Ramp Insertion	
	100 pcm/sec	**200 pcm/s**
Initial Power (MW_t)	0.0001	0.0001
Power Peaking Factor	3.0	3.0
Flow Rate (gpm)	900	900
Core Inlet Temperature (°F)	117	117
Ramp Reactivity Insertion Rate (pcm/second)	100	200
Over-power SCRAM Setpoint (MWt)	2.0	2.0
SCRAM Delay (seconds)	0.05	0.05
Rod Drop Time (seconds/24 inches)	1.0	1.0

[a]Maximum cladding temperature in NUREG 1537 Part1 Appendix 14.1 is listed as 1500 °C (2732 °F) for PULSTAR fuel. The cladding temperature limit of 1200 °C (2200 °F) along with a 17% cladding oxidation limit was set based on discussion with the NRC via teleconference followed up by an email on August 1, 2018.

Table 4-21 – Sequence of Events – Ramp Reactivity Insertion Accident

Parameters	Ramp Insertion	
	Time (sec)	
	100 pcm/sec	200 pcm/sec
Initiation of Ramp Reactivity Insertion	0.0	0.0
Reactor SCRAM Signal	7.41	3.93
SCRAM Rods Start to Drop	7.46	3.99
Control Rods Fully Inserted	8.46	4.99
Maximum Power Level	7.47 (3.32 MWt)	3.98 (4.96 MWt)
Maximum Cladding Temperature	7.91 (124.1 °F)	4.35 (120.7 °F)
Maximum Fuel Temperature	8.48 (161.6 °F)	4.84 (139.5 °F)
Shutdown Core Power at 100 seconds	332 watts	106 watts

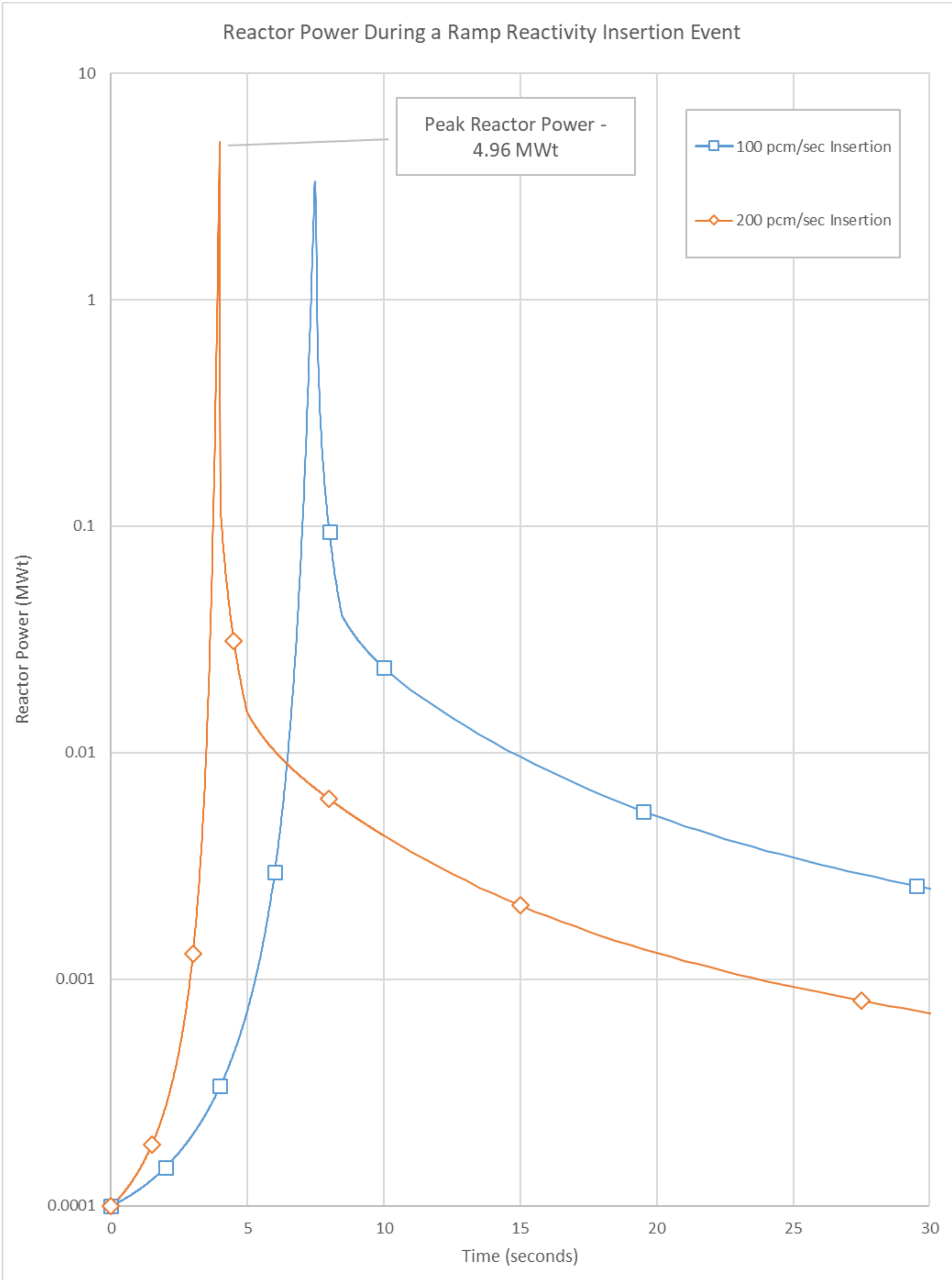


Figure 4-31 – Reactor Power – Ramp Reactivity Insertion Accident

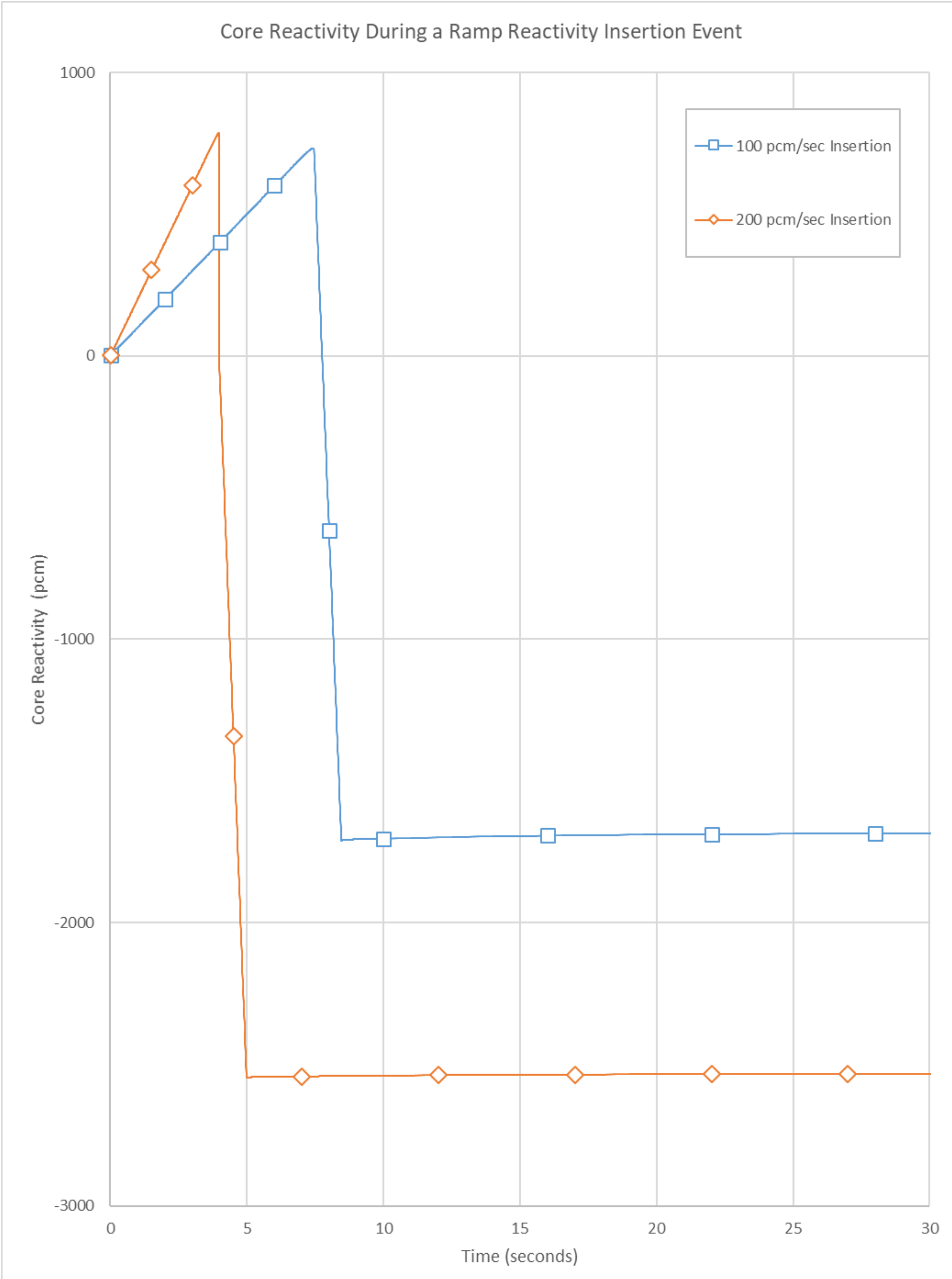


Figure 4-32 – Core Reactivity – Ramp Reactivity Insertion Accident

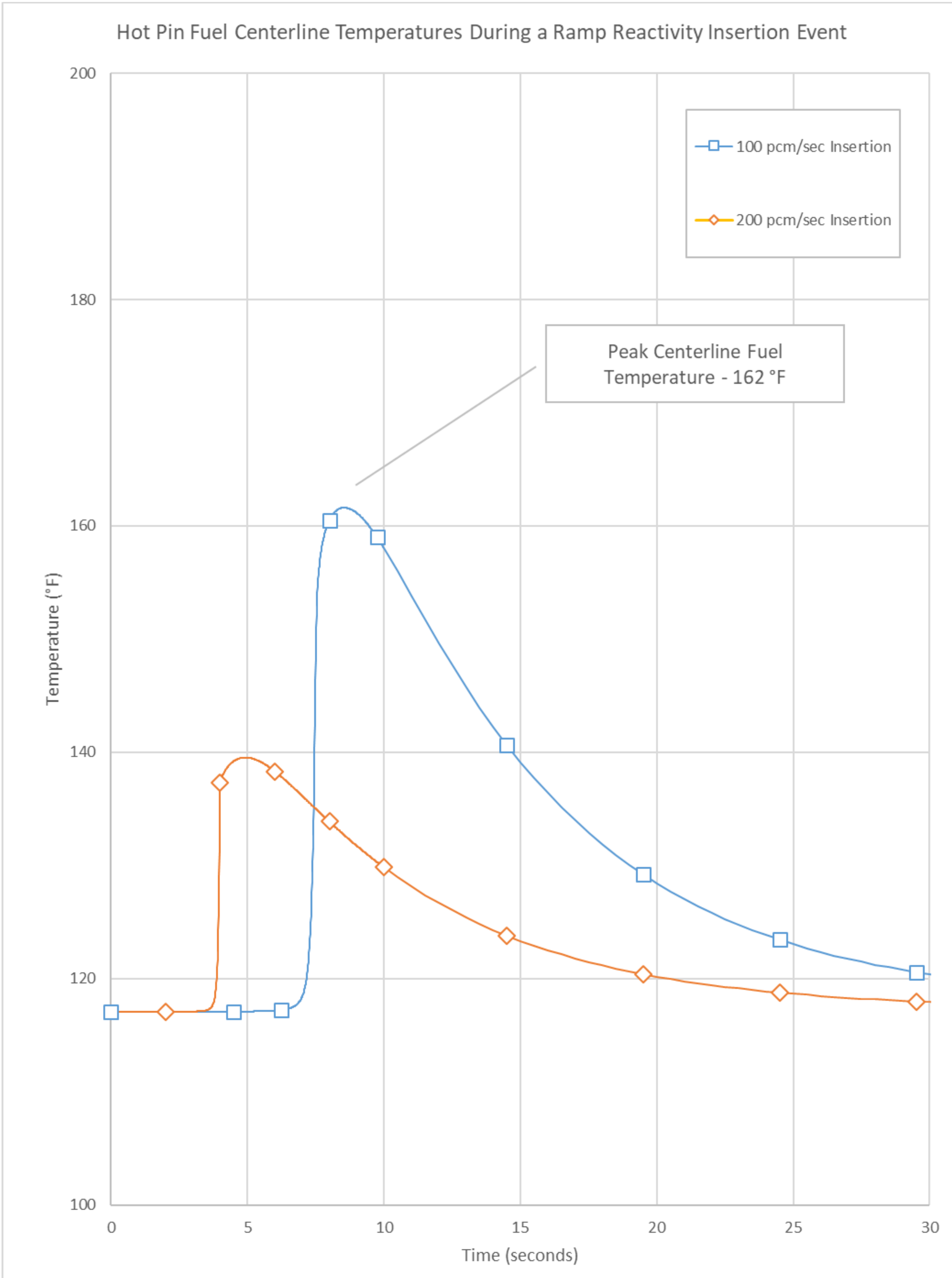


Figure 4-33 – Maximum Fuel Temperature – Ramp Reactivity Insertion Accident

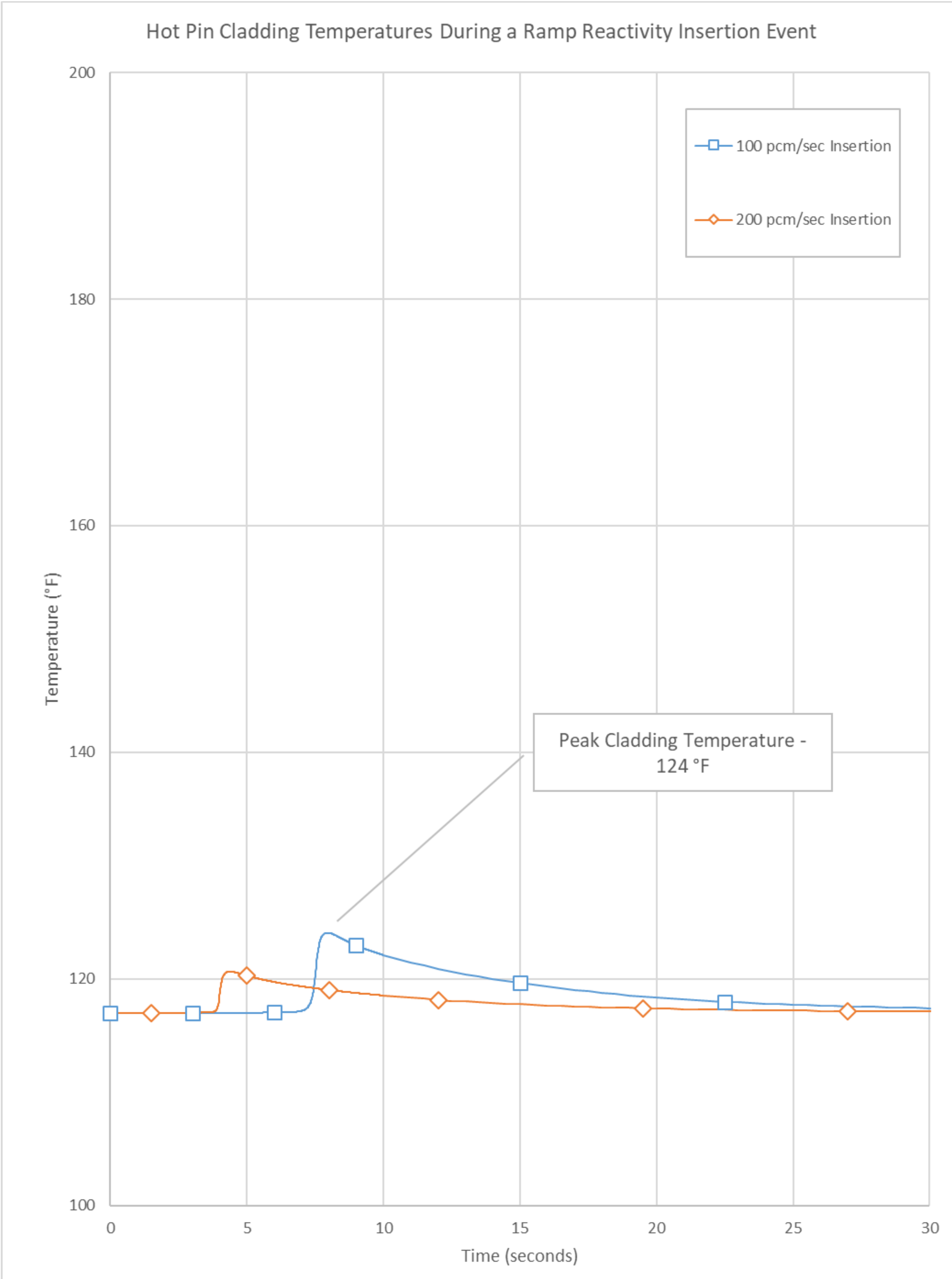


Figure 4-34 – Maximum Cladding Temperature – Ramp Reactivity Insertion Accident

4.3.3.3. Step Reactivity Insertions

The objective is to assess impacts on reactor safety and to ensure the fuel and cladding integrity during a step reactivity insertion, therefore preventing the release of fission products to the environment.

The following parameters are investigated:

- Maximum instantaneous power level (MWt)
- Maximum hot pin fuel centerline temperatures (°F)
- Maximum hot pin cladding temperatures (°F)

Three power/flow scenarios are evaluated:

- Hot zero power without forced flow
- Hot zero power with forced flow
- Hot full power with forced flow

Hot power refers to a primary coolant temperature at the LSSS set-point of 117 °F, which is more limiting than the cold coolant temperature of 70 °F. For this analysis, zero power is defined to be 100 watts and full power refers to 1.85 MWt. The hot full power scenario conservatively initiates at the nominal full power level rather than at the LSSS to allow time for the insertion and resulting transient prior to the initiation of the over-power scram. All other initial conditions are at the LSSS setpoint or Technical Specification limit and are given in Table 4-22.

It is assumed that mishandling of the fuel occurs under the following conditions:

- The core has been loaded in the optimum configuration that places the core and associated parameters at the Technical Specification limits.
- The reactor is critical and operating under the conditions listed in Table 4-22.
- A fuel assembly is dropped from a height of two feet above the core and rapidly enters the core and causes step reactivity insertion of $2.2 in 0.2 seconds.
- The hot pin in the core is assumed to have a peaking factor at the Technical Specification limit of 3.0.

The complete event sequences are summarized in Table 4-23. The step insertion happens at time equal to 0.0 seconds. The scram signal actuates at the scram setpoint of 2.0 MWt with core flow, and 0.25 MWt without core flow. The rods begin to drop after a 0.05 second scram circuitry delay. Control rod drop times are at the Technical Specification limit of 1.0 seconds.

The analysis shows that the power excursions are turned over by the temperature increase and void formations in the core along with the prompt negative temperature coefficient (Doppler) of the fuel. The controls rods continue to insert and are fully inserted in 1.0 seconds and ensure a stable subcritical reactor. Core reactivity during the transient is shown in Figure 4-36, with a peak core net reactivity of 1539 pcm for the zero power transient.

As shown in Figure 4-35 the most limiting events for power excursion are those initiated from zero power and not those initiated from nominal full power. The nominal full power event is terminated more quickly by Doppler feedback and the over-power scram, achieving a maximum power level of 621.7 MWt with a resulting cladding temperature of 333.9 °F.

For the two zero power excursion scenarios, both result in nearly identical maximum powers of 3187 MWt. However, bulk boiling in the core due to the power excursion turns to single phase quickly by the forced cooling in the Hot Zero Power with Flow case, which makes cladding temperature peak at 1722.6 °F around 3.0 seconds. Conversely, in the Hot Zero Power without Flow case, the bulk boiling lasts until the power decreases enough to collapse the boiling, resulting in the fuel cladding temperatures remaining at higher temperatures for a longer period of time and eventually peaking at 1989.5 °F around 7.6 seconds. This is shown in Figure 4-38. Maximum fuel temperatures are 3117.9 °F without flow and 3131.0 °F with forced cooling. Maximum fuel and cladding temperatures are well below the safety limits of 4352 °F and 2200 °F for PULSTAR fuel.[a][3] Fuel and cladding temperatures for the hot pin are shown in Figure 4-37 and Figure 4-38 and are listed in Table 4-23.

Table 4-22– Initialization – Step Reactivity Insertion Accident

Parameters	Step Insertion		
	Hot Full Power with Flow	**Hot Zero Power with Flow**	**Hot Zero Power without Flow**
Reactor Power (MWt)	1.85	0.0001	0.0001
Power Peaking Factor	3.0	3.0	3.0
Primary Flow (gpm)	900	900	0
Core Inlet Temperature (°F)	117	117	117
Secondary Coolant System	ON	OFF	OFF
Secondary Flow (gpm)	1007	N/A	N/A
Secondary Inlet Temperature (°F)	98.8	N/A	N/A
Secondary Pressure (psia)	33	N/A	N/A
Step Reactivity Insertion over 0.2 seconds (pcm)	1612 ($2.2)	1612 ($2.2)	1612($2.2)
SCRAM Setpoint (MWt)	2.0	2.0	0.25
SCRAM Delay (sec)	0.05	0.05	0.05
Control Rod Drop Time (sec)	1.0	1.0	1.0

[a]Maximum cladding temperature in NUREG 1537 Part1 Appendix 14.1 is listed as 1500 °C (2732 °F) for PULSTAR fuel. The cladding temperature limit of 1200 °C (2200 °F) along with a 17% cladding oxidation limit was set based on discussion with the NRC via teleconference followed up by an email on August 1, 2018.

Table 4-23– Sequence of Events – Step Reactivity Insertion Accident

Parameters	Step Insertion		
	Hot Full Power with Flow (sec)	**Hot Zero Power with Flow (sec)**	**Hot Zero Power without Flow (sec)**
Initiation of Step Reactivity Insertion	0.0	0.0	0.0
Reactor SCRAM Signal	0.012	0.17	0.16
SCRAM Rods Start to Drop	0.062	0.22	0.21
End of Reactivity Insertion	0.2	0.2	0.2
Control Rods Fully Inserted	1.062	1.22	1.21
Maximum Power Level (sec)	0.19 (621.7 MWt)	0.21 (3187.3 MWt)	0.21 (3186.3 MWt)
Maximum Cladding Temperature (sec)	0.26 (333.9 °F)	3.0 (1722.6 °F)	7.6 (1989.5 °F)
Maximum Fuel Temperature (sec)	2.0 (2646.6 °F)	3.3 (3131.0 °F)	3.5 (3117.9 °F)
Shutdown Core Power at 100 sec	0.17 MWt	0.04 MWt	0.03 MWt

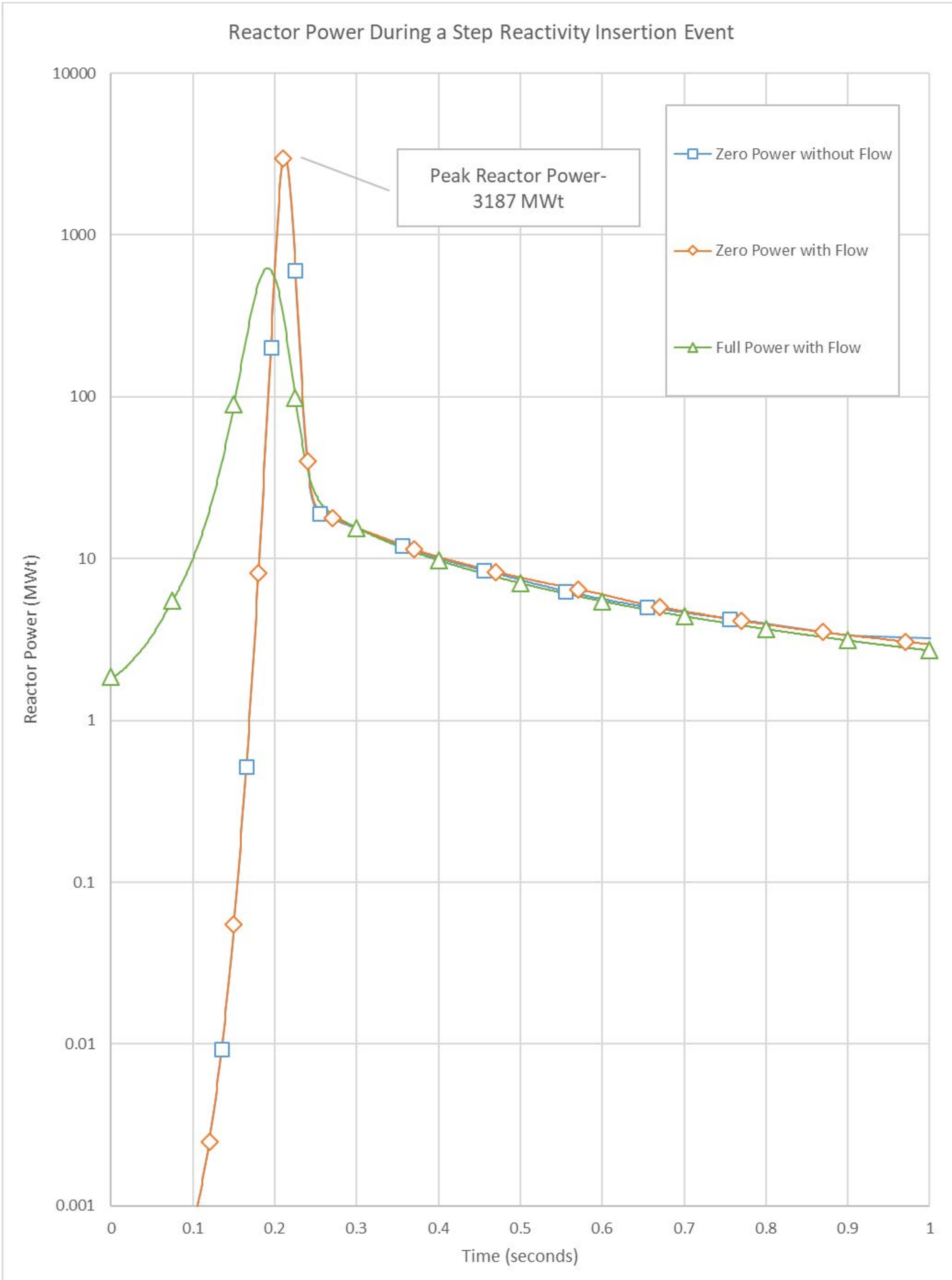


Figure 4-35 – Reactor Power – Step Reactivity Insertion Accident

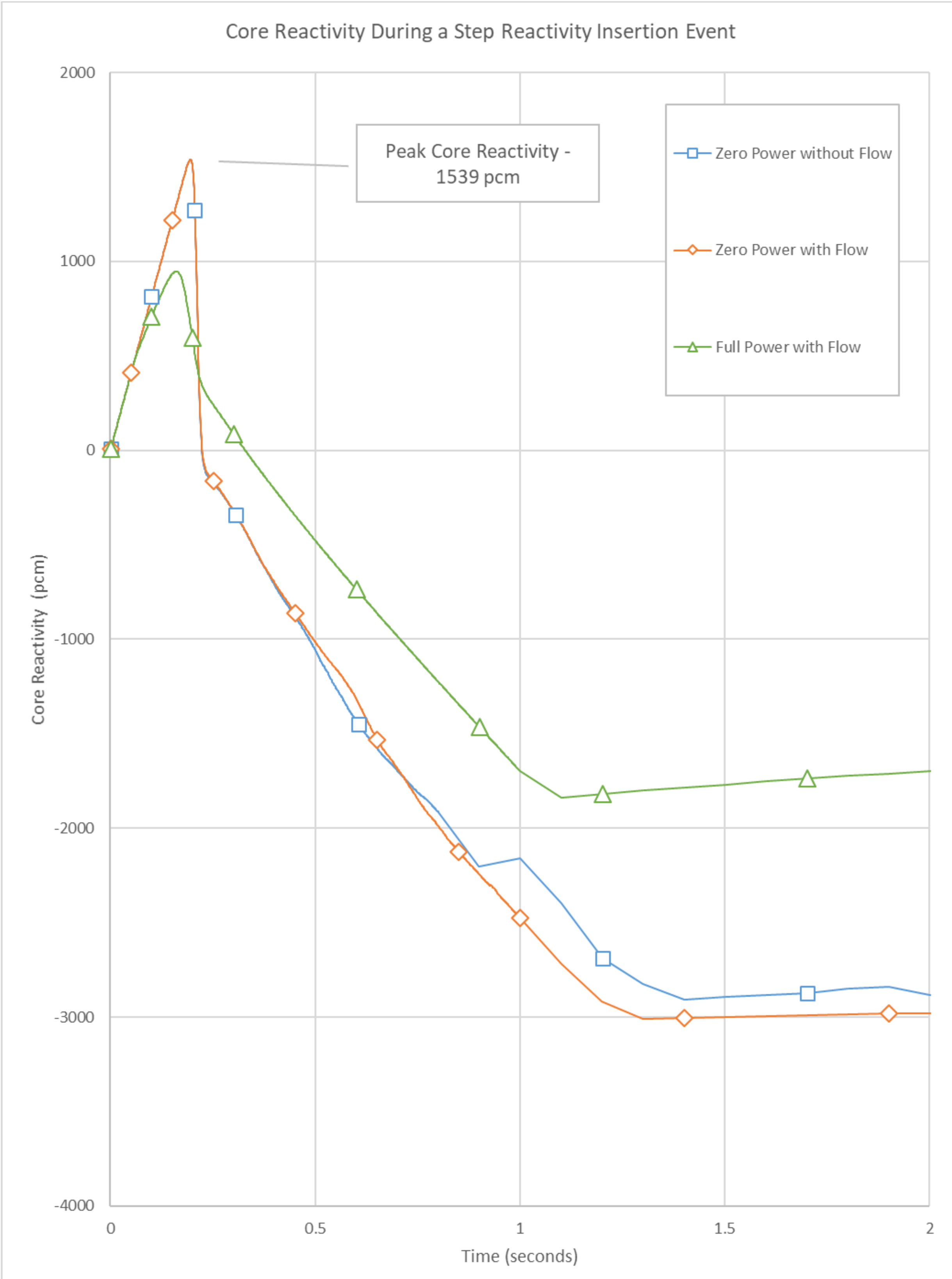


Figure 4-36 – Core Reactivity – Step Reactivity Insertion Accident

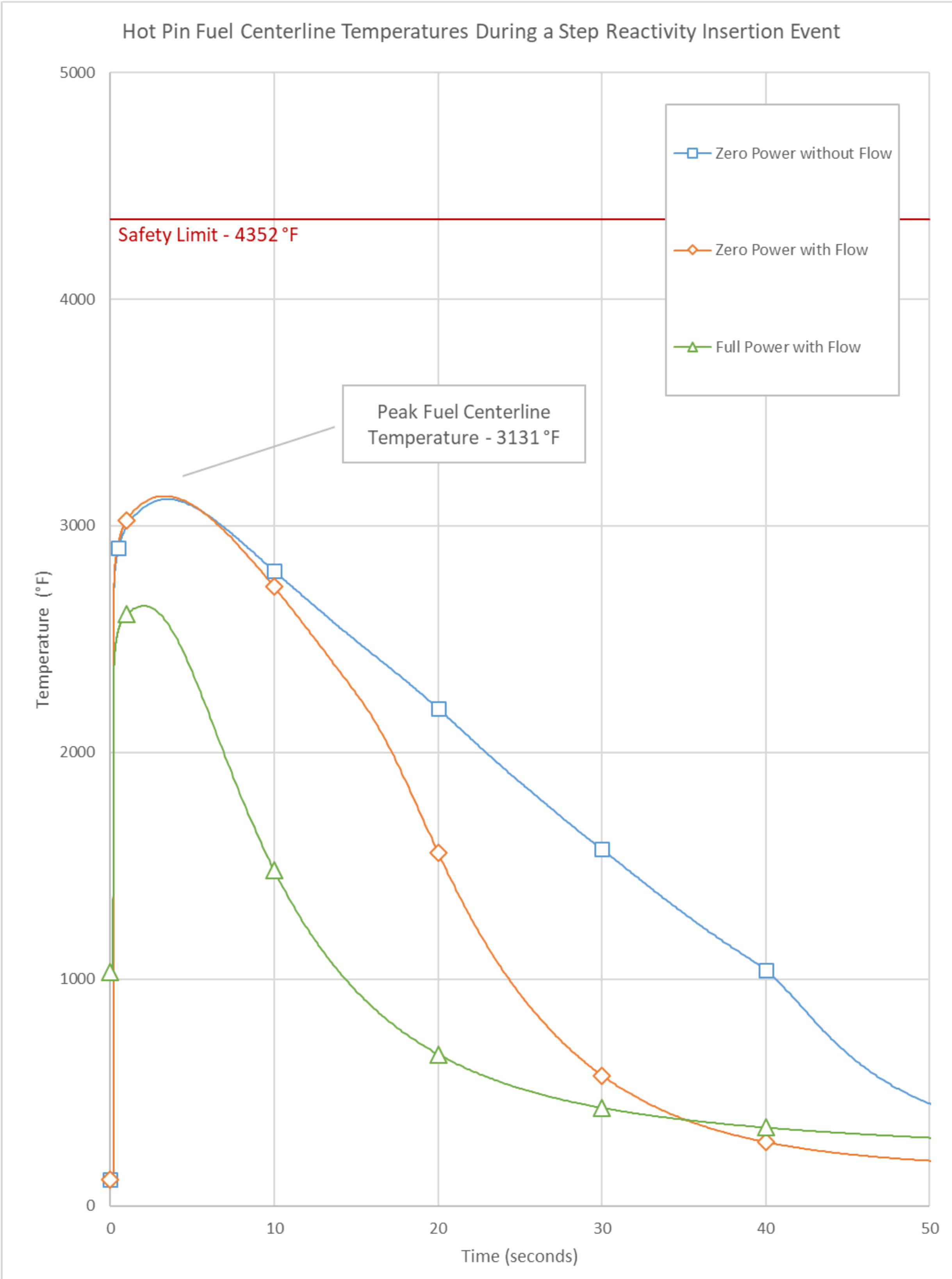


Figure 4-37 – Maximum Fuel Temperature – Step Reactivity Insertion Accident

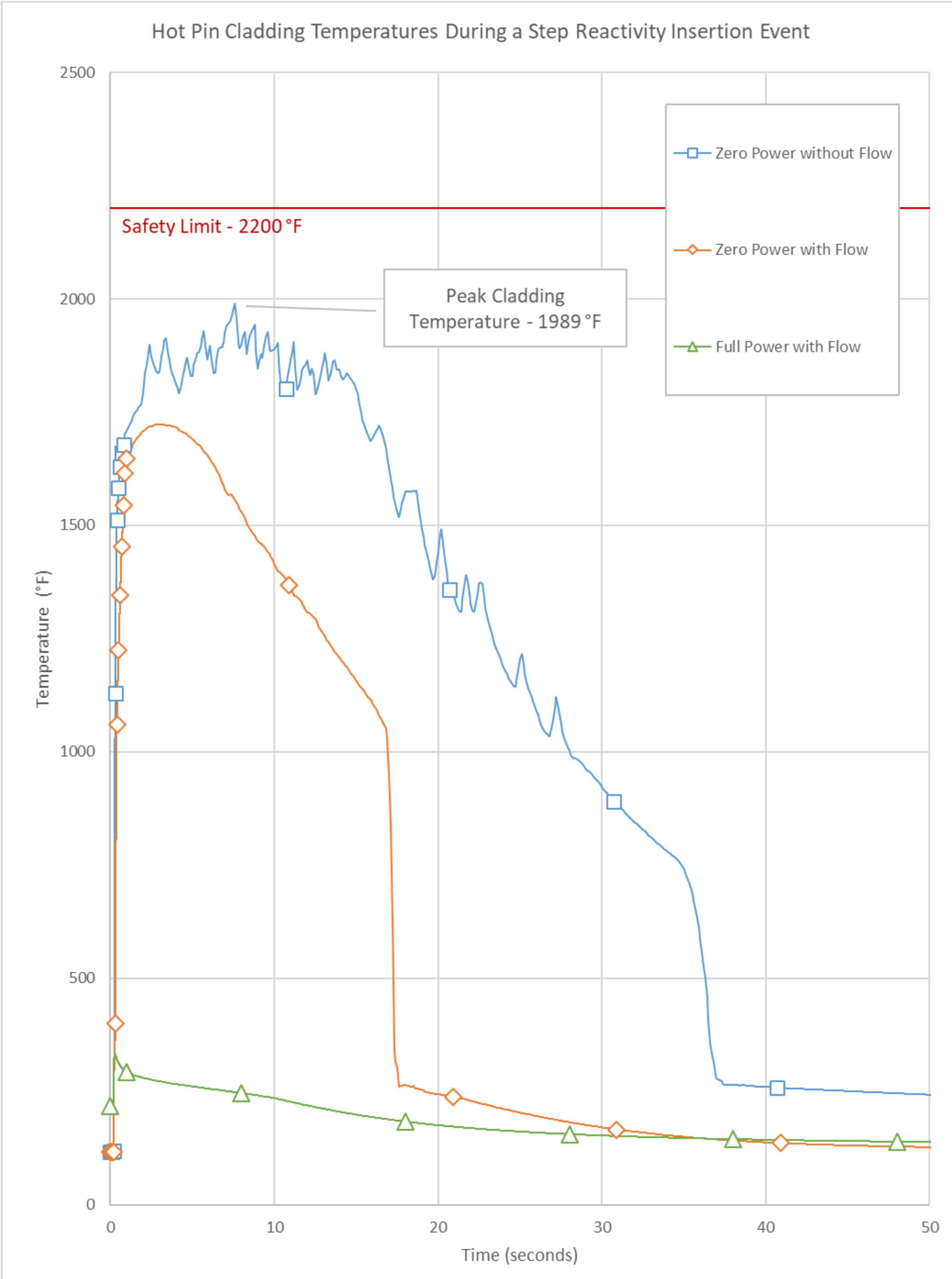


Figure 4-38 – Maximum Cladding Temperature – Step Reactivity Insertion Accident

4.3.3.4. Summary of Reactivity Insertion Accidents

A summary of reactivity insertion accidents is shown in Table 4-24. Ramp reactivity insertions of up to 200 pcm/s pose no hazards to the safety of the reactor. The maximum power achieved and resulting temperatures are well below the safety limits.[a][3]

The most limiting step insertion is the case of hot zero power without forced flow. This is a realistic scenario to consider since fuel is loaded with the primary pump off. Even in this most limiting case the fuel cladding temperatures are well below the limit specified in NUREG 1537. [a][3]

Table 4-24 – Summary – Reactivity Insertion Accidents

Parameter	Ramp Insertion		Step Insertion		
	100 pcm/sec	**200 pcm/sec**	**Hot Nominal Full Power with Flow**	**Hot Zero Power with Flow**	**Hot Zero Power without Flow**
Maximum Power Level (MWt)	3.32	4.96	621.7	3187.3	3186.3
Maximum Fuel Temperature (°F)	161.6	139.5	2646.6	3131.0	3117.9
Maximum Cladding Temperature (°F)	124.1	120.7	333.9	1722.6	1989.5

[a]Maximum cladding temperature in NUREG 1537 Part1 Appendix 14.1 is listed as 1500 °C (2732 °F) for PULSTAR fuel. The cladding temperature limit of 1200 °C (2200 °F) along with a 17% cladding oxidation limit was set based on discussion with the NRC via teleconference followed up by an email on August 1, 2018.

5. Conclusions

The objective of this report was to perform the design analysis and to analyze accident scenarios to evaluate the potential for increasing the PULSTAR nominal steady-state power from 1 MWt to 2 MWt. Steady state operations, loss of primary flow, loss of coolant, and reactivity insertion accidents were all analyzed using the RELAP5/MOD3.3 code of the U.S. Nuclear Regulatory Commission (NRC). [1]

The design calculations performed for the NCSU PULSTAR Reactor determine the limits of operation beyond which the design criteria are violated. The objective of the analyses was to determine the effects of variation in core coolant flow, system pressure, and core coolant inlet temperature on the steady-state burnout level, flow stability, and coolant bulk boiling in the channel of the core having the greatest heat input.

Even though the NCSU PULSTAR operates at only one forced convection flow condition, the safety limit can be established as a function of power and flow and is shown in Figure 4-4. The Limiting Safety System Setting (LSSS) for reactor power has been selected to be 2.0 MWt while the LSSS for reactor coolant flow rate has been selected to be 900 gpm. The selection of these LSSS are sufficient to allow for corrective actions by the safety system to return the reactor to normal operating conditions or to shut the reactor down before a safety limit would be reached. Steady-state thermal margins at 2MWt full power limiting conditions are summarized in Table 4-2, which shows that fuel and cladding temperatures remain well below the safety limits.

For the natural circulation mode of operation, the fuel and cladding temperatures as summarized in Table 4-7 remain well below the safety limits for power levels up to 1.0 MWt. The LSSS for power level of 250 kWt was selected for natural circulation operation due to the potential for elevated dose rates from ^{16}N at the pool top and not due to potential loss of fuel integrity.

Limiting accident scenarios for blocked flow, loss of primary flow, loss of coolant and reactivity insertion accidents were analyzed. Blocked flow and loss of primary coolant flow accidents initiated at the 2.0 MWt limiting condition pose no hazards to the safety of the reactor. As summarized in Table 4-10 and Table 4-14, the maximum resulting fuel and cladding temperatures are well below the safety limits. The loss of coolant accident was analyzed from three different break locations as summarized in Table 4-19. In all scenarios, the failures are assumed to be catastrophic guillotine type breaks while the reactor had been operating at full power for an infinite amount of time. At all times during the accident scenarios, maximum fuel and cladding temperatures remain below the safety limits. The ramp insertions of reactivity of up to 200 pcm/s pose no hazards to the safety of the reactor. Even the most limiting step insertion case of hot zero power without forced flow results in fuel and cladding temperatures, summarized in Table 4-24, that are well below safety limits.

The results of these analyses show that the integrity of the fuel will be maintained for all steady-state and transient limiting accident conditions for reactor power levels up to 2 MWt. The safety margin created by adhering to these limits will prevent the reactor design criteria from being exceeded due to abnormal occurrences.

6. References

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